288

105 1-H2 Y3.N88:10/1202

NUREG-1202

COMPLETED

623

Technical Specifications
Hope Creek Generating Station

Docket No. 50-354

Appendix "A" to
License No. NPF-57

U.S. Nuclear Regulatory
Commission

Office of Nuclear Reactor Regulation

July 1986

ORIGINAL

NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N.W.
   Washington, DC 20555
2. The Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082,
   Washington, DC 20013-7082
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances*.

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

NUREG-1202

Technical Specifications
Hope Creek Generating Station

Docket No. 50-354

Appendix "A" to
License No. NPF-57

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

July 1986

INDEX

SECTION

1.0 DEFINITIONS

PAGE

1.1 ACTION.. 1-1

1.2 AVERAGE PLANAR EXPOSURE.. 1-1

1.3 AVERAGE PLANAR LINEAR HEAT GENERATION RATE.. 1-1

1.4 CHANNEL CALIBRATION.. 1-1

1.5 CHANNEL CHECK.. 1-1

1.6 CHANNEL FUNCTIONAL TEST.. 1-1

1.7 CORE ALTERATION.. 1-2

1.8 CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY.. 1-2

1.9 CRITICAL POWER RATIO.. 1-2

1.10 DOSE EQUIVALENT I-131.. 1-2

1.11 $\overline{E}$-AVERAGE DISINTEGRATION ENERGY.. 1-2

1.12 EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME.. 1-2

1.13 END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME.. 1-3

1.14 FRACTION OF LIMITING POWER DENSITY.. 1-3

1.15 FRACTION OF RATED THERMAL POWER.. 1-3

1.16 FREQUENCY NOTATION.. 1-3

1.17 IDENTIFIED LEAKAGE.. 1-3

1.18 ISOLATION SYSTEM RESPONSE TIME.. 1-3

1.19 LIMITING CONTROL ROD PATTERN.. 1-3

1.20 LINEAR HEAT GENERATION RATE.. 1-4

1.21 LOGIC SYSTEM FUNCTIONAL TEST.. 1-4

1.22 MAXIMUM FRACTION OF LIMITING POWER DENSITY.. 1-4

1.23 MEMBER(S) OF THE PUBLIC.. 1-4

1.24 MINIMUM CRITICAL POWER RATIO.. 1-4

DEFINITIONS

SECTION

DEFINITIONS (Continued) PAGE

1.25 OFF-GAS RADWASTE TREATMENT SYSTEM.................................. 1-4
1.26 OFFSITE DOSE CALCULATION MANUAL.................................. 1-4
1.27 OPERABLE - OPERABILITY.................................. 1-5
1.28 OPERATIONAL CONDITION - CONDITION.................................. 1-5
1.29 PHYSICS TESTS.................................. 1-5
1.30 PRESSURE BOUNDARY LEAKAGE.................................. 1-5
1.31 PRIMARY CONTAINMENT INTEGRITY.................................. 1-5
1.32 PROCESS CONTROL PROGRAM.................................. 1-6
1.33 PURGE-PURGING.................................. 1-6
1.34 RATED THERMAL POWER.................................. 1-6
1.35 REACTOR PROTECTION SYSTEM RESPONSE TIME.................................. 1-6
1.36 REPORTABLE EVENT.................................. 1-6
1.37 ROD DENSITY.................................. 1-6
1.38 SECONDARY CONTAINMENT INTEGRITY.................................. 1-7
1.39 SHUTDOWN MARGIN.................................. 1-7
1.40 SITE BOUNDARY.................................. 1-7
1.41 SOLIDIFICATION.................................. 1-8
1.42 SOURCE CHECK.................................. 1-8
1.43 STAGGERED TEST BASIS.................................. 1-8
1.44 THERMAL POWER.................................. 1-8
1.45 TURBINE BYPASS SYSTEM RESPONSE TIME.................................. 1-8
1.46 UNIDENTIFIED LEAKAGE.................................. 1-8
1.47 UNRESTRICTED AREA.................................. 1-8

SECTION

DEFINITIONS (Continued)

PAGE

1.48 VENTILATION EXHAUST TREATMENT SYSTEM.................................. 1-9

1.49 VENTING.. 1-9

TABLE 1.1, SURVEILLANCE FREQUENCY NOTATION.................................. 1-10

TABLE 1.2, OPERATIONAL CONDITIONS... 1-11

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

SECTION PAGE

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow.............................. 2-1

THERMAL POWER, High Pressure and High Flow.......................... 2-1

Reactor Coolant System Pressure...................................... 2-1

Reactor Vessel Water Level... 2-2

2.2 LIMITING SAFETY SYSTEM SETTINGS

Reactor Protection System Instrumentation Setpoints.................. 2-3

Table 2.2.1-1 Reactor Protection System Instrumentation Setpoints.... 2-4

BASES

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow.............................. B 2-1

THERMAL POWER, High Pressure and High Flow.......................... B 2-2

Table B2.1.2-1 Uncertainties Used in the Determination of the Fuel Cladding Safety Limit.................. B 2-3

Table B2.1.2-2 Nominal Values of Parameters Used in the Statistical Analysis of Fuel Cladding Integrity Safety Limit.................. B 2-4

Reactor Coolant System Pressure...................................... B 2-5

Reactor Vessel Water Level... B 2-5

2.2 LIMITING SAFETY SYSTEM SETTINGS

Reactor Protection System Instrumentation Setpoints.................. B 2-6

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

3/4.0 APPLICABILITY.......... 3/4 0-1

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 SHUTDOWN MARGIN.......... 3/4 1-1

3/4.1.2 REACTIVITY ANOMALIES.......... 3/4 1-2

3/4.1.3 CONTROL RODS

Control Rod Operability.......... 3/4 1-3
Control Rod Maximum Scram Insertion Times.......... 3/4 1-6
Control Rod Average Scram Insertion Times.......... 3/4 1-7
Four Control Rod Group Scram Insertion Times.......... 3/4 1-8
Control Rod Scram Accumulators.......... 3/4 1-9
Control Rod Drive Coupling.......... 3/4 1-11
Control Rod Position Indication.......... 3/4 1-13
Control Rod Drive Housing Support.......... 3/4 1-15

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Rod Worth Minimizer.......... 3/4 1-16
Rod Sequence Control System.......... 3/4 1-17
Rod Block Monitor.......... 3/4 1-18

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM.......... 3/4 1-19

Figure 3.1.5-1 Sodium Pentaborate Solution Volume/ Concentration Requirements.......... 3/4 1-21

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE.......... 3/4 2-1

Maximum Average Planar Linear Heat Generation Rate (MAPHLGR) versus Average Planar Exposure

Figure 3.2.1-1 Initial Core Fuel Type P8CIB071.......... 3/4 2-2

Figure 3.2.1-2 Initial Core Fuel Type P8CIB094.......... 3/4 2-3

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

Figure 3.2.1-3 Initial Core Fuel Type P8CIB163.......... 3/4 2-4
Figure 3.2.1-4 Initial Core Fuel Type P8CIB248.......... 3/4 2-5
Figure 3.2.1-5 Initial Core Fuel Type P8CIB278.......... 3/4 2-6
3/4 2.2 APRM SETPOINTS.. 3/4 2-7
3/4.2.3 MINIMUM CRITICAL POWER RATIO............................ 3/4 2-8
Figure 3.2.3-1 Minimum Critical Power Ratio (MCPR) τ versus at Rated Flow.................... 3/4 2-10
Figure 3.2.3-2 K_f Factor.............................. 3/4 2-11
Table 3.2.3-1 MCPR Feedwater Heating Capacity Adjustment.............................. 3/4 2-12
3/4.2.4 LINEAR HEAT GENERATION RATE............................. 3/4 2-13

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION................ 3/4 3-1
Table 3.3.1-1 Reactor Protection System Instrumentation.......................... 3/4 3-2
Table 3.3.1-2 Reactor Protection System Response Times.................................. 3/4 3-6
Figure 4.3.1.1-1 Reactor Protection System Surveillance Requirements......................... 3/4 3-7
3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION...................... 3/4 3-9
Table 3.3.2-1 Isolation Actuation Instrumentation....... 3/4 3-11
Table 3.3.2-1 Isolation Actuation Instrumentation Setpoints.............................. 3/4 3-22
Table 3.3.2-3 Isolation System Instrumentation Response Time.................................. 3/4 3-26
Table 4.3.2.1-1 Isolation Actuation Instrumentation Surveillance Requirements................ 3/4 3-28

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION... 3/4 3-32

Table 3.3.3-1 Emergency Core Cooling System Actuation Instrumentation........................ 3/4 3-33

Table 3.3.3-2 Emergency Core Cooling System Actuation Instrumentation Setpoints................ 3/4 3-36

Table 3.3.3-3 Emergency Core Cooling System Response Times.................................. 3/4 3-38

Table 4.3.3.1-1 Emergency Core Cooling System Actuation Instrumentation Surveillance Requirements.......................... 3/4 3-39

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS Recirculation Pump Trip System Instrumentation....... 3/4 3-41

Table 3.3.4.1-1 ATWS Recirculation Pump Trip System Instrumentation....................... 3/4 3-42

Table 3.3.4.1-2 ATWS Recirculation Pump Trip System Instrumentation Setpoints.............. 3/4 3-43

Table 4.3.4.1-1 ATWS Recirculation Pump Trip Actuation Instrumentation Surveillance Requirements.......................... 3/4 3-44

End-of-Cycle Recirculation Pump Trip System Instrumentation.. 3/4 3-45

Table 3.3.4.2-1 End-of-Cycle Recirculation Pump Trip System Instrumentation................. 3/4 3-47

Table 3.3.4.2-2 End-of-Cycle Recirculation Pump Trip Setpoints............................ 3/4 3-48

Table 3.3.4.2-3 End-of-Cycle Recirculation Pump Trip System Response Time.................... 3/4 3-49

Table 4.3.4.2.1-1 End-of-Cycle Recirculation Pump Trip System Surveillance Requirements...... 3/4 3-50

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION.. 3/4 3-51

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

Table 3.3.5-1 Reactor Core Isolation Cooling System Actuation Instrumentation................ 3/4 3-52

Table 3.3.5-2 Reactor Core Isolation Cooling System Actuation Instrumentation Setpoints....... 3/4 3-54

Table 4.3.5.1-1 Reactor Core Isolation Cooling System Actuation Instrumentation Surveillance Requirements............................ 3/4 3-55

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION........................ 3/4 3-56

Table 3.3.6-1 Control Rod Block Instrumentation......... 3/4 3-57

Table 3.3.6-2 Control Rod Block Instrumentation Setpoints............................... 3/4 3-59

Table 4.3.6-1 Control Rod Block Instrumentation Surveillance Requirements................ 3/4 3-60

3/4.3.7 MONITORING INSTRUMENTATION

Radiation Monitoring Instrumentation..................... 3/4 3-62

Table 3.3.7.1-1 Radiation Monitoring Instrumentation.... 3/4 3-63

Table 4.3.7.1-1 Radiation Monitoring Instrumentation Surveillance Requirements.............. 3/4 3-66

Seismic Monitoring Instrumentation....................... 3/4 3-68

Table 3.3.7.2-1 Seismic Monitoring Instrumentation...... 3/4 3-69

Table 4.3.7.2-1 Seismic Monitoring Instrumentation Surveillance Requirements.............. 3/4 3-70

Meteorological Monitoring Instrumentation................ 3/4 3-71

Table 3.3.7.3-1 Meteorological Monitoring Instrumentation........................ 3/4 3-72

Table 4.3.7.3-1 Meteorological Monitoring Instrumentation Surveillance Requirements............................ 3/4 3-73

Remote Shutdown Monitoring Instrumentation and Controls... 3/4 3-74

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

Table 3.3.7.4-1 Remote Shutdown Monitoring Instrumentation........................ 3/4 3-75

Table 3.3.7.4-2 Remote Shutdown Systems Controls........ 3/4 3-77

Table 4.3.7.4-1 Remote Shutdown Monitoring Instrumentation Surveillance Requirements........................... 3/4 3-82

Accident Monitoring Instrumentation......................... 3/4 3-84

Table 3.3.7.5-1 Accident Monitoring Instrumentation..... 3/4 3-85

Table 4.3.7.5-1 Accident Monitoring Instrumentation Surveillance Requirements.............. 3/4 3-87

Source Range Monitors.................................... 3/4 3-88

Traversing In-Core Probe System.......................... 3/4 3-89

Loose-Part Detection System.............................. 3/4 3-90

Radioactive Liquid Effluent Monitoring Instrumentation.... 3/4 3-91

Table 3.3.7.9-1 Radioactive Liquid Effluent Monitoring Instrumentation............. 3/4 3-92

Table 4.3.7.9-1 Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements.......................... 3/4 3-94

Radioactive Gaseous Effluent Monitoring Instrumentation... 3/4 3-96

Table 3.3.7.10-1 Radioactive Gaseous Effluent Monitoring Instrumentation............ 3/4 3-97

Table 4.3.7.10-1 Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements......................... 3/4 3-100

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM..................... 3/4 3-103

3/4.3.9 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION.. 3/4 3-105

Table 3.3.9-1 Feedwater/Main Turbine Trip System Actuation Instrumentation................ 3/4 3-106

BEST COPY [illegible]

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

Table 3.3.9-2 Feedwater/Main Turbine Trip System Actuation Instrumentation Setpoints....... 3/4 3-107

Table 4.3.9.1-1 Feedwater/Main Turbine Trip System Actuation Instrumentation Surveillance Requirement............................ 3/4 3-108

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

Recirculation Loops.. 3/4 4-1

Figure 3.4.1.1-1 % Rated Thermal Power Versus Core Flow........................... 3/4 4-3

Jet Pumps.. 3/4 4-4

Recirculation Pumps.. 3/4 4-5

Idle Recirculation Loop Startup............................ 3/4 4-6

3/4.4.2 SAFETY/RELIEF VALVES

Safety/Relief Valves....................................... 3/4 4-7

Safety/Relief Valves Low-Low Set Function................. 3/4 4-9

3/4 4.3 REACTOR COOLANT SYSTEM LEAKAGE

Leakage Detection Systems.................................. 3/4 4-10

Operational Leakage.. 3/4 4-11

Table 3.4.3.1-1 Reactor Coolant System Pressure Isolation Valves........................ 3/4 4-13

Table 3.4.3.2-2 Reactor Coolant System Interface Valves Leakage Pressure Monitors....... 3/4 4-14

3/4.4.4 CHEMISTRY... 3/4 4-15

Table 3.4.4-1 Reactor Coolant System Chemistry Limits................................. 3/4 4-17

3/4.4.5 SPECIFIC ACTIVITY.. 3/4 4-18

Table 4.4.5-1 Primary Coolant Specific Activity Sample and Analysis Program..................... 3/4 4-20

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

Reactor Coolant System.................................. 3/4 4-21

Figure 3.4.6.1-1 Minimum Reactor Pressure Vessel Metal Temperature Versus Reactor Vessel Pressure............................ 3/4 4-23

Table 4.4.6.1.3-1 Reactor Vessel Material Surveillance Program Withdrawal Schedule.......... 3/4 4-24

Reactor Steam Dome...................................... 3/4 4-25

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES.......................... 3/4 4-26

3/4.4.8 STRUCTURAL INTEGRITY.. 3/4 4-27

3/4.4.9 RESIDUAL HEAT REMOVAL

Hot Shutdown.. 3/4 4-28

Cold Shutdown... 3/4 4-29

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ECCS - OPERATING.. 3/4 5-1

3/4.5.2 ECCS - SHUTDOWN... 3/4 5-6

3/4.5.3 SUPPRESSION CHAMBER....................................... 3/4 5-8

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

Primary Containment Integrity............................ 3/4 6-1

Primary Containment Leakage............................. 3/4 6-2

Primary Containment Air Locks........................... 3/4 6-5

MSIV Sealing System...................................... 3/4 6-7

Primary Containment Structural Integrity................. 3/4 6-8

Drywell and Suppression Chamber Internal Pressure......... 3/4 6-9

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

Drywell Average Air Temperature.......................... 3/4 6-10
Drywell and Suppression Chamber Purge System............. 3/4 6-11

3/4.6.2 DEPRESSURIZATION SYSTEMS

Suppression Chamber...................................... 3/4 6-12
Suppression Pool Spray................................... 3/4 6-15
Suppression Pool Cooling................................. 3/4 6-16

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES...................... 3/4 6-17

Table 3.6.3-1 Primary Containment Isolation Valves...... 3/4 6-19

3/4.6.4 VACUUM RELIEF

Suppression Chamber - Drywell Vacuum Breakers............ 3/4 6-43
Reactor Building - Suppression Chamber Vacuum Breakers....................................... 3/4 6-45

3/4.6.5 SECONDARY CONTAINMENT

Secondary Containment Integrity.......................... 3/4 6-47
Secondary Containment Automatic Isolation Dampers......... 3/4 6-49

Table 3.6.5.2-1 Secondary Containment Ventilation System Automatic Isolation Dampers Isolation Group No. 19................. 3/4 6-50

Filtration, Recirculation and Ventilation System.......... 3/4 6-51

3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL

Containment Hydrogen Recombiner Systems.................. 3/4 6-54
Drywell and Suppression Chamber Oxygen Concentration...... 3/4 6-55

3/4.7 PLANT SYSTEMS

3/4.7.1 SERVICE WATER SYSTEMS

Safety Auxiliaries Cooling System........................ 3/4 7-1
Station Service Water System............................. 3/4 7-3
Ultimate Heat Sink....................................... 3/4 7-5

3/4.7.2 CONTROL ROOM EMERGENCY FILTRATION SYSTEM................. 3/4 7-6

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

3/4.7.3 FLOOD PROTECTION.................................... 3/4 7-9

Table 3.7.3-1 Perimeter Flood Doors.................... 3/4 7-10

3/4.7.4 REACTOR CORE ISOLATION COOLING SYSTEM.................... 3/4 7-11

3/4.7.5 SNUBBERS.................................... 3/4 7-13

Figure 4.7.5-1 Sample Plan 2) for Snubber Functional Test.................................... 3/4 7-18

3/4.7.6 SEALED SOURCE CONTAMINATION.................................... 3/4 7-19

3.4.7.7 MAIN TURBINE BYPASS SYSTEM.................................... 3/4 7-21

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

A.C. Sources-Operating.................................... 3/4 8-1

Table 4.8.1.1.2-1 Diesel Generator Test Schedule........ 3/4 8-10

A.C. Sources-Shutdown.................................... 3/4 8-11

3/4.8.2 D.C. SOURCES

D.C. Sources-Operating.................................... 3/4 8-12

Table 4.8.2.1-1 Battery Surveillance Requirements........ 3/4 8-16

D.C. Sources-Shutdown.................................... 3/4 8-17

3/4.8.3 ONSITE POWER DISTRIBUTION SYSTEMS

Distribution - Operating.................................... 3/4 8-18

Distribution - Shutdown.................................... 3/4 8-21

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Primary Containment Penetration Conductor Overcurrent Protective Devices.................................... 3/4 8-24

Table 3.8.4.1-1 Primary Containment Penetration Conductor Overcurrent Protective Devices.......... 3/4 8-26

Motor Operated Valve Thermal Overload Protection (Bypassed).................................... 3/4 8-30

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

Table 3.8.4.2-1 Motor Operated Valves-Thermal Overload Protection (Bypassed).................. 3/4 8-31

Motor Operated Valve Thermal Overload Protection (Not Bypassed).. 3/4 8-38

Table 3.8.4.3-1 Motor Operated Valves-Thermal Overload Protection (Not Bypassed)............... 3/4 8-39

Reactor Protection System Electric Power Monitoring....... 3/4 8-40

Class 1E Isolation Breaker Overcurrent Protection Devices (Breaker Tripped by LOCA Signal)............... 3/4 8-41

Table 3.8.4.5-1 Class 1E Isolation Breaker Overcurrent Protective Devices (Breaker Tripped by a LOCA Signal)......................... 3/4 8-42

Power Range Neutron Monitoring System Electric Power Monitoring.. 3/4 8-44

3/4.9 REFUELING OPERATIONS

3/4.9.1 REACTOR MODE SWITCH.................................. 3/4 9-1

3/4.9.2 INSTRUMENTATION.. 3/4 9-3

3/4.9.3 CONTROL ROD POSITION................................. 3/4 9-5

3/4.9.4 DECAY TIME.. 3/4 9-6

3/4.9.5 COMMUNICATIONS... 3/4 9-7

3/4.9.6 REFUELING PLATFORM..................................... 3/4 9-8

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL.................... 3/4 9-10

3/4.9.8 WATER LEVEL - REACTOR VESSEL............................. 3/4 9-11

3/4.9.9 WATER LEVEL - SPENT FUEL STORAGE POOL..................... 3/4 9-12

3/4.9.10 CONTROL ROD REMOVAL

Single Control Rod Removal................................. 3/4 9-13

Multiple Control Rod Removal............................... 3/4 9-15

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

High Water Level.................................... 3/4 9-17
Low Water Level..................................... 3/4 9-18

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY............................ 3/4 10-1
3/4.10.2 ROD SEQUENCE CONTROL SYSTEM.............................. 3/4 10-2
3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS........................... 3/4 10-3
3/4.10.4 RECIRCULATION LOOPS...................................... 3/4 10-4
3/4.10.5 OXYGEN CONCENTRATION..................................... 3/4 10-5
3/4.10.6 TRAINING STARTUPS.. 3/4 10-6
3/4.10.7 SPECIAL INSTRUMENTATION - INITIAL CORE LOADING........... 3/4 10-7

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

Concentration.. 3/4 11-1
Table 4.11.1.1.1-1 Radioactive Liquid Waste Sampling and Analysis Program.... 3/4 11-2
Dose... 3/4 11-5
Liquid Waste Treatment................................... 3/4 11-6
Liquid Holdup Tanks...................................... 3/4 11-7

3/4.11.2 GASEOUS EFFLUENTS

Dose Rate.. 3/4 11-8
Table 4.11.2.1.2-1 Radioactive Gaseous Waste Sampling and Analysis Program.... 3/4 11-9
Dose - Noble Gases....................................... 3/4 11-12
Dose - Iodine-131, Iodine-133, Tritium, and Radionuclides in Particulate Form.... 3/4 11-13
Gaseous Radwaste Treatment............................... 3/4 11-14
Ventilation Exhaust Treatment System..................... 3/4 11-15

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

Explosive Gas Mixture.. 3/4 11-16
Main Condenser... 3/4 11-17
Venting or Purging... 3/4 11-18

3/4.11.3 SOLID RADIOACTIVE WASTE TREATMENT........................ 3/4 11-19

3/4.11.4 TOTAL DOSE... 3/4 11-20

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM....................................... 3/4 12-1

Table 3.12.1-1 Radiological Environmental Monitoring Program.................. 3/4 12-3

Table 3.12.1-2 Reporting Levels For Radioactivity Concentrations In Environmental Samples............................ 3/4 12-9

Table 4.12.1-1 Detection Capabilities For Environmental Sample Analysis........ 3/4 12-10

3/4.12.2 LAND USE CENSUS.. 3/4 12-13

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM........................ 3/4 12-14

INDEX

BASES

SECTION PAGE

3/4.0 APPLICABILITY.. B 3/4 0-1

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 SHUTDOWN MARGIN.. B 3/4 1-1
3/4.1.2 REACTIVITY ANOMALIES.. B 3/4 1-1
3/4.1.3 CONTROL RODS.. B 3/4 1-2
3/4.1.4 CONTROL ROD PROGRAM CONTROLS.. B 3/4 1-3
3/4.1.5 STANDBY LIQUID CONTROL SYSTEM.. B 3/4 1-4

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE........... B 3/4 2-1
3/4.2.2 APRM SETPOINTS.. B 3/4 2-2
Table B3.2.1-1 Significant Input Parameters to the Loss-of-Coolant Accident Analysis.................... B 3/4 2-3
3/4.2.3 MINIMUM CRITICAL POWER RATIO........................ B 3/4 2-4
3/4.2.4 LINEAR HEAT GENERATION RATE........................ B 3/4 2-5

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION........... B 3/4 3-1
3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION................. B 3/4 3-2
3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION.. B 3/4 3-2
3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION.... B 3/4 3-3
3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION.. B 3/4 3-4
3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION................... B 3/4 3-4
3/4.3.7 MONITORING INSTRUMENTATION
Radiation Monitoring Instrumentation................. B 3/4 3-4

INDEX

BASES

SECTION PAGE

INSTRUMENTATION (Continued)

Seismic Monitoring Instrumentation .... B 3/4 3-4
Meteorological Monitoring Instrumentation .... B 3/4 3-4
Remote Shutdown Monitoring Instrumentation and Controls .... B 3/4 3-5
Accident Monitoring Instrumentation .... B 3/4 3-5
Source Range Monitors .... B 3/4 3-5
Traversing In-Core Probe System .... B 3/4 3-5
Loose-Part Detection System .... B 3/4 3-6
Radioactive Liquid Effluent Monitoring Instrumentation .... B 3/4 3-6
Radioactive Gaseous Effluent Monitoring Instrumentation .... B 3/4 3-6

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM .... B 3/4 3-7

3/4.3.9 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION .... B 3/4 3-7

Figure B3/4 3-1 Reactor Vessel Water Level .... B 3/4 3-8

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM .... B 3/4 4-1

3/4.4.2 SAFETY/RELIEF VALVES .... B 3/4 4-2

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

Leakage Detection Systems .... B 3/4 4-3
Operational Leakage .... B 3/4 4-3

3/4.4.4 CHEMISTRY .... B 3/4 4-3

3/4.4.5 SPECIFIC ACTIVITY .... B 3/4 4-4

3/4.4.6 PRESSURE/TEMPERATURE LIMITS .... B 3/4 4-5

Table B3/4.4.6-1 Reactor Vessel Toughness .... B 3/4 4-7

Figure B3/4.4.6-1 Fast Neutron Fluence (E>1Mev) at (1/4)T as a Function of Service Life .... B 3/4 4-8

SECTION PAGE

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES.................... B 3/4 4-6
3/4.4.8 STRUCTURAL INTEGRITY.................................. B 3/4 4-6
3/4.4.9 RESIDUAL HEAT REMOVAL................................ B 3/4 4-6

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1/2 ECCS - OPERATING and SHUTDOWN.......................... B 3/4 5-1
3/4.5.3 SUPPRESSION CHAMBER.................................. B 3/4 5-2

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

Primary Containment Integrity....................... B 3/4 6-1
Primary Containment Leakage......................... B 3/4 6-1
Primary Containment Air Locks....................... B 3/4 6-1
MSIV Sealing System................................. B 3/4 6-1
Primary Containment Structural Integrity............ B 3/4 6-2
Drywell and Suppression Chamber Internal Pressure.... B 3/4 6-2
Drywell Average Air Temperature..................... B 3/4 6-2
Drywell and Suppression Chamber Purge System......... B 3/4 6-2
3/4.6.2 DEPRESSURIZATION SYSTEMS............................. B 3/4 6-3
3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES................. B 3/4 6-5
3/4.6.4 VACUUM RELIEF.. B 3/4 6-5
3/4.6.5 SECONDARY CONTAINMENT................................ B 3/4 6-5
3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL............... B 3/4 6-6

BASES

SECTION PAGE

3/4.7 PLANT SYSTEMS

3/4.7.1 SERVICE WATER SYSTEMS.................................. B 3/4 7-1

3/4.7.2 CONTROL ROOM EMERGENCY FILTRATION SYSTEM............. B 3/4 7-1

3/4.7.3 FLOOD PROTECTION...................................... B 3/4 7-1

3/4.7.4 REACTOR CORE ISOLATION COOLING SYSTEM................. B 3/4 7-1

3/4.7.5 SNUBBERS... B 3/4 7-2

3/4.7.6 SEALED SOURCE CONTAMINATION........................... B 3/4 7-4

3/4.7.7 MAIN TURBINE BYPASS SYSTEM............................ B 3/4 7-4

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1, 3/4.8.2 and
3/4.8.3 A.C. SOURCES, D.C. SOURCES and ONSITE POWER
DISTRIBUTION SYSTEMS................................... B 3/4 8-1

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES................ B 3/4 8-3

3/4.9 REFUELING OPERATIONS

3/4.9.1 REACTOR MODE SWITCH.................................... B 3/4 9-1

3/4.9.2 INSTRUMENTATION.. B 3/4 9-1

3/4.9.3 CONTROL ROD POSITION.................................. B 3/4 9-1

3/4.9.4 DECAY TIME... B 3/4 9-1

3/4.9.5 COMMUNICATIONS... B 3/4 9-1

3/4.9.6 REFUELING PLATFORM..................................... B 3/4 9-2

3/4.9.7 CRANE TRAVEL-SPENT FUEL STORAGE POOL.................. B 3/4 9-2

3/4.9.8 and 3/4.9.9 WATER LEVEL - REACTOR VESSEL
and WATER LEVEL - SPENT FUEL STORAGE POOL............ B 3/4 9-2

3/4.9.10 CONTROL ROD REMOVAL.................................... B 3/4 9-2

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION........ B 3/4 9-2

INDEX

BASES

SECTION PAGE

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY........................ B 3/4 10-1
3/4.10.2 ROD SEQUENCE CONTROL SYSTEM.......................... B 3/4 10-1
3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS....................... B 3/4 10-1
3/4.10.4 RECIRCULATION LOOPS.................................. B 3/4 10-1
3/4.10.5 OXYGEN CONCENTRATION................................. B 3/4 10-1
3/4.10.6 TRAINING STARTUPS.................................... B 3/4 10-1
3/4.10.7 SPECIAL INSTRUMENTATION - INITIAL CORE LOADING....... B 3/4 10-1

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

Concentration.. B 3/4 11-1
Dose... B 3/4 11-1
Liquid Radwaste Treatment System............................... B 3/4 11-2
Liquid Holdup Tanks.. B 3/4 11-2

3/4.11.2 GASEOUS EFFLUENTS

Dose Rate.. B 3/4 11-2
Dose - Noble Gases... B 3/4 11-3
Dose - Iodine-131, Iodine-133, Tritium, and Radionuclides in Particulate Form..................... B 3/4 11-3
Gaseous Radwaste Treatment System and Ventilation Exhaust Treatment Systems.................. B 3/4 11-4
Explosive Gas Mixture.. B 3/4 11-4
Main Condenser... B 3/4 11-5
Venting or Purging... B 3/4 11-5
3/4.11.3 SOLID RADIOACTIVE WASTE TREATMENT.................... B 3/4 11-5
3/4.11.4 TOTAL DOSE... B 3/4 11-5

BASES

SECTION PAGE

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM.. B 3/4 12-1
3/4.12.2 LAND USE CENSUS.. B 3/4 12-1
3/4.12.3 INTERLABORATORY COMPARISON PROGRAM........................ B 3/4 12-2

INDEX

DESIGN FEATURES

SECTION PAGE

5.1 SITE

Exclusion Area and Map Defining Unrestricted Area and Site Boundary for Radioactive Gaseous and Liquid Effluents.......... 5-1

Figure 5.1.1-1 Exclusion Area and Unrestricted Areas and Site Boundary for Radioactive Gaseous and Liquid Effluents........................ 5-2

Low Population Zone... 5-1

Figure 5.1.2-1 Low Population Zone...................... 5-3

5.2 CONTAINMENT

Configuration.. 5-1

Design Temperature and Pressure............................. 5-1

Secondary Containment....................................... 5-1

5.3 REACTOR CORE

Fuel Assemblies... 5-4

Control Rod Assemblies...................................... 5-4

5.4 REACTOR COOLANT SYSTEM

Design Pressure and Temperature............................. 5-4

Volume... 5-4

5.5 METEOROLOGICAL TOWER LOCATION............................ 5-4

5.6 FUEL STORAGE

Criticality.. 5-5

Drainage... 5-5

Capacity... 5-5

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT........................ 5-5

Table 5.7.1-1 Component Cyclic or Transient Limits........ 5-6

INDEX

ADMINISTRATIVE CONTROLS

SECTION PAGE

6.1 RESPONSIBILITY 6-1

6.2 ORGANIZATION 6-1

6.2.1 OFFSITE 6-1

6.2.2 UNIT STAFF 6-1

Figure 6.2.1-1 Offsite Organization 6-3

Figure 6.2.2-1 Unit Organization 6-4

Figure 6.2.2-2 Minimum Shift Crew Composition Single Unit Facility 6-5

6.2.3 SHIFT TECHNICAL ADVISOR 6-6

6.3 UNIT STAFF QUALIFICATIONS 6-6

6.4 TRAINING 6-6

6.5 REVIEW AND AUDIT 6-6

6.5.1 STATION OPERATIONS REVIEW COMMITTEE (SORC) 6-6

FUNCTION 6-6

COMPOSITION 6-7

ALTERNATES 6-7

MEETING FREQUENCY 6-7

QUORUM 6-7

RESPONSIBILITIES 6-7

REVIEW PROCESS 6-8

AUTHORITY 6-9

RECORDS 6-9

SECTION PAGE

6.5.2 NUCLEAR SAFETY REVIEW (NSR)................................ 6-9
FUNCTION.. 6-9
COMPOSITION... 6-9
CONSULTANTS... 6-10
OFFSITE SAFETY REVIEW (OSR)................................ 6-10
FUNCTION.. 6-10
REVIEW.. 6-10
AUDITS.. 6-11
RECORDS... 6-12
ONSITE SAFETY REVIEW GROUP (SRG)........................... 6-12
RESPONSIBILITIES.. 6-12
AUTHORITY... 6-13
6.5.3 TECHNICAL REVIEW AND CONTROL.............................. 6-13
ACTIVITIES.. 6-13
PROCEDURE RELATED DOCUMENTS............................... 6-13
NON-PROCEDURE RELATED DOCUMENTS........................... 6-14
RECORDS... 6-14
6.6 REPORTable EVENT ACTION.................................. 6-14
6.7 SAFETY LIMIT VIOLATION................................... 6-14
6.8 PROCEDURES AND PROGRAMS.................................. 6-15
6.9 REPORTING REQUIREMENTS................................... 6-17
6.9.1 ROUTINE REPORTS.. 6-17
STARTUP REPORT.. 6-17

ADMINISTRATIVE CONTROLS

SECTION PAGE

ANNUAL REPORTS.. 6-17

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT........ 6-18

SEMIANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT.... 6-19

MONTHLY OPERATING REPORTS.................................. 6-20

6.9.2 SPECIAL REPORTS.. 6-20

6.10 RECORD RETENTION.. 6-21

6.11 RADIATION PROTECTION PROGRAM.............................. 6-22

6.12 HIGH RADIATION AREA.. 6-22

6.13 PROCESS CONTROL PROGRAM (PCP).............................. 6-23

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)..................... 6-24

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS.............................. 6-24

1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AVERAGE PLANAR EXPOSURE

1.2 The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is tested.

CORE ALTERATION

1.7 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Normal movement of the SRMs, IRMs, TIPs, or special movable detectors is not considered a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY

1.8 The CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD) shall be highest value of the FLPD which exists in the core.

CRITICAL POWER RATIO

1.9 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the GEXL correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

1.12 The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function, i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.13 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker from initial movement of the associated:

a. Turbine stop valves, and

b. Turbine control valves.

The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

FRACTION OF LIMITING POWER DENSITY

1.14 The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.

FRACTION OF RATED THERMAL POWER

1.15 The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.

FREQUENCY NOTATION

1.16 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

IDENTIFIED LEAKAGE

1.17 IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

1.18 The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

LIMITING CONTROL ROD PATTERN

1.19 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

1.20 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

1.21 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e., all relays and contacts, all trip units, solid state logic elements, etc, of a logic circuit, from sensor through and including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

MAXIMUM FRACTION OF LIMITING POWER DENSITY

1.22 The MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) shall be highest value of the FLPD which exists in the core.

MEMBER(S) OF THE PUBLIC

1.23 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, it contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

MINIMUM CRITICAL POWER RATIO

1.24 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

OFF-GAS RADWASTE TREATMENT SYSTEM

1.25 An OFF-GAS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting reactor coolant system offgases from the main condenser evacuation system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

OFFSITE DOSE CALCULATION MANUAL

1.26 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the radiological environmental monitoring program.

OPERABLE - OPERABILITY

1.27 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION - CONDITION

1.28 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

PHYSICS TESTS

1.29 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.30 PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

1.31 PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.3-1 of Specification 3.6.3.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. The primary containment leakage rates are within the limits of Specification 3.6.1.2.

e. The suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM

1.32 The PROCESS CONTROL PROGRAM (PCP) shall contain the provisions to assure that the SOLIDIFICATION or dewatering and packaging of radioactive wastes results in a waste package with properties that meet the minimum and stability requirements of 10 CFR Part 61 and other requirements for transportation to the disposal site and receipt at the disposal site. With SOLIDIFICATION, the PCP shall identify the process parameters influencing SOLIDIFICATION such as pH, oil content, H_2O content, solids content ratio of solidification agent to waste and/or necessary additives for each type of anticipated waste, and the acceptable boundary conditions for the process parameters shall be identified for each waste type, based on laboratory scale and full scale testing or experience. With dewatering, the PCP shall include an identification of conditions that must be satisfied, based on full scale testing, to assure that dewatering of bead resins, powdered resins, and filter sludges will result in volumes of free water, at the time of disposal, within the limits of 10 CFR Part 61 and of the low-level radioactive waste disposal site.

PURGE - PURGING

1.33 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.34 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3293 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.35 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REPORTABLE EVENT

1.36 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.37 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

1.38 SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All secondary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve or damper, as applicable secured in its closed position, except as provided in Table 3.6.5.2-1 of Specification 3.6.5.2.

b. All secondary containment hatches and blowout panels are closed and sealed.

c. The filtration, recirculation and ventilation system is in compliance with the requirements of Specification 3.6.5.3.

d. For double door arrangements, at least one door in each access to the secondary containment is closed.

e. For single door arrangements, the door in each access to the secondary containment is closed, except for normal entry and exit.

f. The sealing mechanism associated with each secondary containment penetration, e.g., welds, bellows or O-rings, is OPERABLE.

g. The pressure within the secondary containment is less than or equal to the value required by Specification 4.6.5.1.a.

SHUTDOWN MARGIN

1.39 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

SITE BOUNDARY

1.40 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled, by the licensee.

SOLIDIFICATION

1.41 SOLIDIFICATION shall be the immobilization of wet radioactive wastes such as evaporator bottoms, spent resins, sludges, and reverse osmosis concentrates as a result of a process of thoroughly mixing the water type with a solidification agent(s) to form a free standing monolith with chemical and physical characteristics specified in the PROCESS CONTROL PROGRAM (PCP).

SOURCE CHECK

1.42 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.43 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals.

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

THERMAL POWER

1.44 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TURBINE BYPASS SYSTEM RESPONSE TIME

1.45 The TURBINE BYPASS SYSTEM RESPONSE TIME consists of two separate time intervals: a) time from initial movement of the main turbine stop valve or control valve until 80% of the turbine bypass capacity is established, and b) the time from initial movement of the main turbine stop valve or control valve until initial movement of the turbine bypass valve. Either response time may be measured by any series of sequential, overlapping, or total steps such that the entire response time is measured.

UNIDENTIFIED LEAKAGE

1.46 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

UNRESTRICTED AREA

1.47 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.48 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.49 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

TABLE 1.1

SURVEILLANCE FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
A	At least once per 366 days.
R	At least once per 18 months (550 days).
S/U	Prior to each reactor startup.
P	Prior to each radioactive release.
N.A.	Not applicable.

TABLE 1.2

OPERATIONAL CONDITIONS

CONDITION	MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE
1. POWER OPERATION	Run	Any temperature
2. STARTUP	Startup/Hot Standby	Any temperature
3. HOT SHUTDOWN	Shutdown#,***	> 200°F
4. COLD SHUTDOWN	Shutdown#,##,***	≤ 200°F
5. REFUELING*	Shutdown or Refuel**,#	≤ 140°F

#The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions and related instrumentation provided that the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

##The reactor mode switch may be placed in the Refuel position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.9.10.1.

*Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

**See Special Test Exceptions 3.10.1 and 3.10.3.

***The reactor mode switch may be placed in the Refuel position while a single control rod is being recoupled or withdrawn provided that the one-rod-out interlock is OPERABLE.

NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Section 2.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.06 with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.06 and the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

13

SAFETY LIMITS (Continued)

REACTOR VESSEL WATER LEVEL

2.1.4 The reactor vessel water level shall be above the top of the active irradiated fuel.

APPLICABILITY: OPERATIONAL CONDITIONS 3, 4 and 5

ACTION:

With the reactor vessel water level at or below the top of the active irradiated fuel, manually initiate the ECCS to restore the water level, after depressurizing the reactor vessel, if required. Comply with the requirements of Specification 6.7.1.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor protection system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2.1-1.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

With a reactor protection system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2.1-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

15

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-Upscale		
1) Flow Biased	≤ 0.66 W+51%, with a maximum of	≤ 0.66 W+54%, with a maximum of
2) High Flow Clamped	≤ 113.5% of RATED THERMAL POWER	≤ 115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux-Upscale	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
d. Inoperative	NA	NA
e. Downscale	≥ 4% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High	≤ 1037 psig	≤ 1057 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 12.5 inches above instrument zero*	≥ 11.0 inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed
6. Main Steam Line Radiation - High, High	≤ 3.0 x full power background	≤ 3.6 x full power background

---

*See Bases Figure B 3/4 3-1.

16

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS
(continued)

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
7. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
8. Scram Discharge Volume Water Level - High		
a. Float Switch	Elevation 110' 10.5"	Elevation 111' 0.5"
b. Level Transmitter/Trip Unit	Elevation 110' 10.5"*	Elevation 111' 4.5"*
9. Turbine Stop Valve - Closure	≤ 5% closed	≤ 7% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 530 psig	≥ 465 psig
11. Reactor Mode Switch Shutdown Position	NA	NA
12. Manual Scram	NA	NA

---

*80.5" above instrument zero EL 104' 2" for Level Transmitter/Trip Unit A&B (South Header)
83.25" above instrument zero EL 103' 11.25" for Level Transmitter/Trip Unit C&D (North Header)

BASES
FOR
SECTION 2.0
SAFETY LIMITS
AND
LIMITING SAFETY SYSTEM SETTINGS

SECTION 2.0

SAFETY LIMITS
AND
LIMITING SAFETY SYSTEM SETTINGS

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than 1.06. MCPR greater than 1.06 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the GEXL correlation is not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

20

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB[a], which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) Boiling Length (L), (GEXL), correlation.

The GEXL correlation is valid over the range of conditions used in the tests of the data used to develop the correlation.

The required input to the statistical model are the uncertainties listed in Bases Table B2.1.2-1 and the nominal values of the core parameters listed in Bases Table B2.1.2-2.

The bases for the uncertainties in the core parameters are given in NEDO-20340[b] and the basis for the uncertainty in the GEXL correlation is given in NEDO-10958-A[a]. The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution during any fuel cycle would not be as severe as the distribution used in the analysis.

---

[a] "General Electric BWR Thermal Analysis Bases (GETAB) Data, Correlation and Design Application," NEDO-10958-A.

[b] General Electric "Process Computer Performance Evaluation Accuracy" NEDO-20340 and Amendment 1, NEDO-20340-1 dated June 1974 and December 1974, respectively.

21

Bases Table B2.1.2-1

UNCERTAINTIES USED IN THE DETERMINATION OF THE FUEL CLADDING SAFETY LIMIT*

Quantity	Standard Deviation (% of Point)
Feedwater Flow	1.76
Feedwater Temperature	0.76
Reactor Pressure	0.5
Core Inlet Temperature	0.2
Core Total Flow	2.5
Channel Flow Area	3.0
Friction Factor Multiplier	10.0
Channel Friction Factor Multiplier	5.0
TIP Readings	6.3
R Factor	1.5
Critical Power	3.6

---

* The uncertainty analysis used to establish the core wide Safety Limit MCPR is based on the assumption of quadrant power symmetry for the reactor core.

22

Bases Table B2.1.2-2

NOMINAL VALUES OF PARAMETERS USED IN THE STATISTICAL ANALYSIS OF FUEL CLADDING INTEGRITY SAFETY LIMIT

THERMAL POWER	3323 MW
Core Flow	108.5 Mlb/hr
Dome Pressure	1010.4 psig
Channel Flow Area	0.1089 ft^2
R-Factor	High enrichment - 1.043 Medium enrichment - 1.039 Low enrichment - 1.030

2.1.3 REACTOR COOLANT SYSTEM PRESSURE

The Safety Limit for the reactor coolant system pressure has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to Section III of the ASME Boiler and Pressure Vessel Code 1968 Edition, including Addenda through Winter 1969, which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The Safety Limit of 1325 psig, as measured by the reactor vassel steam dome pressure indicator, is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The reactor coolant system is designed to the USAS Nuclear Power Piping Code, Section B31.7 1969 Edition, including Addenda through July 1, 1970 for the reactor recirculation piping, which permits a maximum pressure transient of 110%, 1375 psig, of design pressure, 1250 psig for suction piping and 1500 psig for discharge piping. The pressure Safety Limit is selected to be the lowest transient overpressure allowed by the applicable codes.

2.1.4 REACTOR VESSEL WATER LEVEL

With fuel in the reactor vessel during periods when the reactor is shutdown, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level became less than two-thirds of the core height. The Safety Limit has been established at the top of the active irradiated fuel to provide a point which can be monitored and also provide adequate margin for effective action.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System instrumentation setpoints specified in Table 2.2.1-1 are the values at which the reactor trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

1. Intermediate Range Monitor, Neutron Flux - High

The IRM system consists of 8 chambers, 4 in each of the reactor trip systems. The IRM is a 5 decade 10 range instrument. The trip setpoint of 120 divisions of scale is active in each of the 10 ranges. Thus as the IRM is ranged up to accommodate the increase in power level, the trip setpoint is also ranged up. The IRM instruments provide for overlap with both the APRM and SRM systems.

The most significant source of reactivity changes during the power increase is due to control rod withdrawal. In order to ensure that the IRM provides the required protection, a range of rod withdrawal accidents have been analyzed. The results of these analyses are in Section 15.4 of the FSAR. The most severe case involves an initial condition in which THERMAL POWER is at approximately 1% of RATED THERMAL POWER. Additional conservatism was taken in this analysis by assuming the IRM channel closest to the control rod being withdrawn is bypassed. The results of this analysis show that the reactor is shutdown and peak power is limited to 21% of RATED THERMAL POWER with the peak fuel enthalpy well below the fuel failure threshold of 170 cal/gm. Based on this analysis, the IRM provides protection against local control rod errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

2. Average Power Range Monitor

For operation at low pressure and low flow during STARTUP, the APRM scram setting of 15% of RATED THERMAL POWER provides adequate thermal margin between the setpoint and the Safety Limits. The margin accommodates the anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor and cold water from sources available during startup is not much colder than that already in the system. Temperature coefficients are small and control rod patterns are constrained by the RSCS and RWM. Of all the possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power increase.

25

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks and because several rods must be moved to change power by a significant amount, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the trip level, the rate of power rise is not more than 5% of RATED THERMAL POWER per minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% neutron flux trip remains active until the mode switch is placed in the Run position.

The APRM trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and therefore the monitors respond directly and quickly to changes due to transient operation for the case of the Fixed Neutron Flux-Upscale setpoint; i.e, for a power increase, the THERMAL POWER of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer associated with the fuel. For the Flow Biased Simulated Thermal Power-Upscale setpoint, a time constant of 6 ± 0.6 seconds is introduced into the flow biased APRM in order to simulate the fuel thermal transient characteristics. A more conservative maximum value is used for the flow biased setpoint as shown in Table 2.2.1-1.

The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown. The flow referenced trip setpoint must be adjusted by the specified formula in Specification 3.2.2 in order to maintain these margins when CMFLPD is greater than or equal to FRTP.

3. Reactor Vessel Steam Dome Pressure-High

High pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating will also tend to increase the power of the reactor by compressing voids thus adding reactivity. The trip will quickly reduce the neutron flux, counteracting the pressure increase. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement compared to the highest pressure that occurs in the system during a transient. This trip setpoint is effective at low power/flow conditions when the turbine control valve fast closure and turbine stop valve closure trip are bypassed. For a load rejection or turbine trip under these conditions, the transient analysis indicated an adequate margin to the thermal hydraulic limit.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

4. Reactor Vessel Water Level-Low

The reactor vessel water level trip setpoint has been used in transient analyses dealing with coolant inventory decrease. The scram setting was chosen far enough below the normal operating level to avoid spurious trips but high enough above the fuel to assure that there is adequate protection for the fuel and pressure limits.

5. Main Steam Line Isolation Valve-Closure

The main steam line isolation valve closure trip was provided to limit the amount of fission product release for certain postulated events. The MSIV's are closed automatically from measured parameters such as high steam flow, high steam line radiation, low reactor water level, high steam tunnel temperature, and low steam line pressure. The MSIV's closure scram anticipates the pressure and flux transients which could follow MSIV closure and thereby protects reactor vessel pressure and fuel thermal/hydraulic Safety Limits.

6. Main Steam Line Radiation-High

The main steam line radiation detectors are provided to detect a gross failure of the fuel cladding. When the high radiation is detected, a trip is initiated to reduce the continued failure of fuel cladding. At the same time the main steam line isolation valves are closed to limit the release of fission products. The trip setting is high enough above background radiation levels to prevent spurious trips yet low enough to promptly detect gross failures in the fuel cladding.

7. Drywell Pressure-High

High pressure in the drywell could indicate a break in the primary pressure boundary systems or a loss of drywell cooling. The reactor is tripped in order to minimize the possibility of fuel damage and reduce the amount of energy being added to the coolant and the primary containment. The trip setting was selected as low as possible without causing spurious trips.

8. Scram Discharge Volume Water Level-High

The scram discharge volume receives the water displaced by the motion of the control rod drive pistons during a reactor scram. Should this volume fill up to a point where there is insufficient volume to accept the displaced water at pressures below 65 psig, control rod insertion would be hindered. The reactor is therefore tripped when the water level has reached a point high enough to indicate that it is indeed filling up, but the volume is still great enough to accommodate the water from the movement of the rods at pressures below 65 psig when they are tripped. The trip setpoint for each scram discharge volume is equivalent to a contained volume of approximately 35 gallons of water.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

9. Turbine Stop Valve-Closure

The turbine stop valve closure trip anticipates the pressure, neutron flux, and heat flux increases that would result from closure of the stop valves. With a trip setting of 5% of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained during the worst case transient.

10. Turbine Control Valve Fast Closure, Trip Oil Pressure-Low

The turbine control valve fast closure trip anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection with or without coincident failure of the turbine bypass valves. The Reactor Protection System initiates a trip when fast closure of the control valves is initiated by the fast acting solenoid valves and in less than 30 milliseconds after the start of control valve fast closure. This is achieved by the action of the fast acting solenoid valves in rapidly reducing hydraulic trip oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the Reactor Protection System. This trip setting, a slower closure time, and a different valve characteristic from that of the turbine stop valve, combine to produce transients which are very similar to that for the stop valve. Relevant transient analyses are discussed in Section 15.2.2 of the Final Safety Analysis Report.

11. Reactor Mode Switch Shutdown Position

The reactor mode switch Shutdown position provides additional manual reactor trip capability.

12. Manual Scram

The Manual Scram pushbutton switches provide a diverse means for initiating a reactor shutdown (scram) to the automatic protective instrumentation channels and provides manual reactor trip capability.

28

SECTIONS 3.0 and 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the OPERATIONAL CONDITIONS or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the Action requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour action shall be initiated to place the unit in an OPERATIONAL CONDITION in which the Specification does not apply by placing it, as applicable, in:

1. At least STARTUP within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

This Specification is not applicable in OPERATIONAL CONDITIONS 4 or 5.

3.0.4 Entry into an OPERATIONAL CONDITION or other specified condition shall not be made unless the conditions for the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL CONDITIONS as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual Specifications.

30

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, but

b. The combined time interval for any 3 consecutive surveillance intervals shall not exceed 3.25 times the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specificatons. Surveillance requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL CONDITION or other specified applicable condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, & 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g) (6) (i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1 The SHUTDOWN MARGIN shall be equal to or greater than:

a. 0.38% delta k/k with the highest worth rod analytically determined, or

b. 0.28% delta k/k with the highest worth rod determined by test.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

a. In OPERATIONAL CONDITION 1 or 2, reestablish the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4, immediately verify all insertable control rods to be inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

c. In OPERATIONAL CONDITION 5, suspend CORE ALTERATIONS and other activities that could reduce the SHUTDOWN MARGIN and insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.1 The SHUTDOWN MARGIN shall be determined to be equal to or greater than specified at any time during the fuel cycle:

a. By measurement, prior to or during the first startup after each refueling.

b. By measurement, within 500 MWD/T prior to the core average exposure at which the predicted SHUTDOWN MARGIN, including uncertainties and calculation biases, is equal to the specified limit.

c. Within 12 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or is untrippable, except that the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod.

33

3/4.1.2 REACTIVITY ANOMALIES

LIMITING CONDITION FOR OPERATION

3.1.2 The reactivity equivalence of the difference between the actual ROD DENSITY and the predicted ROD DENSITY shall not exceed 1% delta k/k.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the reactivity equivalence difference exceeding 1% delta k/k:

a. Within 12 hours perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected.

b. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.2 The reactivity equivalence of the difference between the actual ROD DENSITY and the predicted ROD DENSITY shall be verified to be less than or equal to 1% delta k/k:

a. During the first startup following CORE ALTERATIONS, and

b. At least once per 31 effective full power days during POWER OPERATION.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 CONTROL RODS

CONTROL ROD OPERABILITY

LIMITING CONDITION FOR OPERATION

3.1.3.1 All control rods shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one control rod inoperable due to being immovable, as a result of excessive friction or mechanical interference, or known to be untrippable:

    1. Within one hour:

        a) Verify that the inoperable control rod, if withdrawn, is separated from all other inoperable control rods by at least two control cells in all directions.

        b) Disarm the associated directional control valves** hydraulically by closing the drive water and exhaust water isolation valves.

        Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

    2. Restore the inoperable control rod, if withdrawn, to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. With one or more control rods trippable but inoperable for causes other than addressed in ACTION a, above:

    1. If the inoperable control rod(s) is withdrawn, within one hour:

        a) Verify that the inoperable withdrawn control rod(s) is separated from all other inoperable withdrawn control rods by at least two control cells in all directions, and

        b) Demonstrate the insertion capability of the inoperable withdrawn control rod(s) by inserting the control rod(s) at least one notch by drive water pressure within the normal operating range*

        Otherwise, insert the inoperable withdrawn control rod(s) and disarm the associated directional control valves** either:

        a) Electrically, or
        b) Hydraulically by closing the drive water and exhaust water isolation valves.

---

*The inoperable control rod may then be withdrawn to a position no further withdrawn than its position when found to be inoperable.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

2. If the inoperable control rod(s) is inserted, within one hour disarm the associated directional control valves** either:

   a) Electrically, or

   b) Hydraulically by closing the drive water and exhaust water isolation valves.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

3. The provisions of Specification 3.0.4 are not applicable.

c. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

d. With one scram discharge volume vent valve and/or one scram discharge volume drain valve inoperable and open, restore the inoperable valve(s) to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours.

e. With any scram discharge volume vent valve(s) and/or any scram discharge volume drain valve(s) otherwise inoperable, restore the inoperable valve(s) to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The scram discharge volume drain and vent valves shall be demonstrated OPERABLE by:

a. At least once per 24 hours verifying each valve to be open,* and

b. At least once per 31 days cycling each valve through at least one complete cycle of full travel.

4.1.3.1.2 When above the low power setpoint of the RWM and RSCS, all withdrawn control rods not required to have their directional control valves disarmed

---

*These valves may be closed intermittently for testing under administrative controls.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

36

electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

a. At least once per 7 days, and

b. At least once per 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference.

4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2, 4.1.3.4, 4.1.3.5, 4.1.3.6 and 4.1.3.7.

4.1.3.1.4 The scram discharge volume shall be determined OPERABLE by demonstrating:

a. The scram discharge volume drain and vent valves OPERABLE at least once per 18 months, by verifying that the drain and vent valves:

    1. Close within 30 seconds after receipt of a signal for control rods to scram, and

    2. Open when the scram signal is reset.

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 5, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the maximum scram insertion time of one or more control rods exceeding 7.0 seconds:

    1. Declare the control rod(s) with the slow insertion time inoperable, and

    2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of 7.0 seconds.

    Otherwise, be in at least HOT SHUTDOWN within 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days.

b. For specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and

c. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION.

CONTROL ROD AVERAGE SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.3 The average scram insertion time of all OPERABLE control rods from the fully withdrawn position, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
45	0.43
39	0.86
25	1.93
05	3.49

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the average scram insertion time exceeding any of the above limits, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.3 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

39

FOUR CONTROL ROD GROUP SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.4 The average scram insertion time, from the fully withdrawn position, for the three fastest control rods in each group of four control rods arranged in a two-by-two array, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
45	0.45
39	0.92
25	2.05
05	3.70

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the average scram insertion times of control rods exceeding the above limits:

   1. Declare the control rods with the slower than average scram insertion times inoperable until an analysis is performed to determine that required scram reactivity remains for the slow four control rod group, and

   2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with an average scram insertion time(s) in excess of the average scram insertion time limit.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.3.4 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

CONTROL ROD SCRAM ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.1.3.5 All control rod scram accumulators shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITIONS 1 or 2:

   1. With one control rod scram accumulator inoperable, within 8 hours:

      a) Restore the inoperable accumulator to OPERABLE status, or

      b) Declare the control rod associated with the inoperable accumulator inoperable.

      Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

   2. With more than one control rod scram accumulator inoperable, declare the associated control rods inoperable and:

      a) If the control rod associated with any inoperable scram accumulator is withdrawn, immediately verify that at least one control rod drive pump is operating by inserting at least one withdrawn control rod at least one notch. If no control rod drive pump is operating: 1) If reactor pressure is $\geq$ 900 psig, restart at least one control rod drive pump within 20 minutes or place the reactor mode switch in the Shutdown position. 2) If reactor pressure is < 900 psig, place the reactor mode switch in the Shutdown position.

      b) Insert the inoperable control rods and disarm the associated control valves either:

         1) Electrically, or

         2) Hydraulically by closing the drive water and exhaust water isolation valves.

      Otherwise, be in at least HOT SHUTDOWN within 12 hours.

b. In OPERATIONAL CONDITION 5*:

   1. With one withdrawn control rod with its associated scram accumulator inoperable, insert the affected control rod and disarm the associated directional control valves within one hour, either:

      a) Electrically, or

      b) Hydraulically by closing the drive water and exhaust water isolation valves.

   2. With more than one withdrawn control rod with the associated scram accumulator inoperable and no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

c. The provisions of Specification 3.0.4 are not applicable.

---

*At least the accumulator associated with each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each control rod scram accumulator shall be determined OPERABLE:

a. At least once per 7 days by verifying that the indicated pressure is greater than or equal to 940 psig unless the control rod is inserted and disarmed or scrammed.

b. At least once per 18 months by:

1. Performance of a:

a) CHANNEL FUNCTIONAL TEST of the leak detectors, and

b) CHANNEL CALIBRATION of the pressure detectors, and verifying an alarm setpoint of 940 + 95, -0 psig on decreasing pressure.

CONTROL ROD DRIVE COUPLING

LIMITING CONDITION FOR OPERATION

3.1.3.6 All control rods shall be coupled to their drive mechanisms.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1 and 2 with one control rod not coupled to its associated drive mechanism, within 2 hours:

    1. If permitted by the RWM and RSCS, insert the control rod to accomplish recoupling and verify recoupling by withdrawing the control rod, and:

        a) Observing any indicated response of the nuclear instrumentation, and

        b) Demonstrating that the control rod will not go to the overtravel position.

    2. If recoupling is not accomplished on the first attempt or, if not permitted by the RWM or RSCS, then until permitted by the RWM and RSCS, declare the control rod inoperable, insert the control rod and disarm the associated directional control valves** either:

        a) Electrically, or

        b) Hydraulically by closing the drive water and exhaust water isolation valves.

    Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5* with a withdrawn control rod not coupled to its associated drive mechanism, within 2 hours either:

    1. Insert the control rod to accomplish recoupling and verify recoupling by withdrawing the control rod and demonstrating that the control rod will not go to the overtravel position, or

    2. If recoupling is not accomplished, insert the control rod and disarm the associated directional control valves** either:

        a) Electrically, or

        b) Hydraulically by closing the drive water and exhaust water isolation valves.

c. The provisions of Specification 3.0.4 are not applicable.

---

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

43

SURVEILLANCE REQUIREMENTS

4.1.3.6 Each affected control rod shall be demonstrated to be coupled to its drive mechanism by observing any indicated response of the nuclear instrumentation while withdrawing the control rod to the fully withdrawn position and then verifying that the control rod drive does not go to the overtravel position:

a. Prior to reactor criticality after completing CORE ALTERATIONS that could have affected the control rod drive coupling integrity,

b. Anytime the control rod is withdrawn to the "Full out" position in subsequent operation, and

c. Following maintenance on or modification to the control rod or control rod drive system which could have affected the control rod drive coupling integrity.

CONTROL ROD POSITION INDICATION

LIMITING CONDITION FOR OPERATION

3.1.3.7 The control rod position indication system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1 or 2 with one or more control rod position indicators inoperable, within 1 hour:

   1. Determine the position of the control rod by utilizing the RSCS substitute position display within preset power level, or:

      a) Moving the control rod, by single notch movement, to a position with an OPERABLE position indicator,

      b) Returning the control rod, by single notch movement, to its original position, and

      c) Verifying no control rod drift alarm at least once per 12 hours, or

   2. Move the control rod to a position with an OPERABLE position indicator, or

   3. When THERMAL POWER is:

      a) Within the preset power level of the RSCS, declare the control rod inoperable.

      b) Greater than the preset power level of the RSCS, declare the control rod inoperable, insert the control rod and disarm the associated directional control valves** either:

         1) Electrically, or

         2) Hydraulically by closing the drive water and exhaust water isolation valves.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5* with a withdrawn control rod position indicator inoperable, move the control rod to a position with an OPERABLE position indicator or insert the control rod.

c. The provisions of Specification 3.0.4 are not applicable.

---

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

45

SURVEILLANCE REQUIREMENTS

4.1.3.7 The control rod position indication system shall be determined OPERABLE by verifying:

a. At least once per 24 hours that the position of each control rod is indicated,

b. That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.1.3.1.2, and

c. That the control rod position indicator corresponds to the control rod position indicated by the "Full Out" position indicator when performing Surveillance Requirement 4.1.3.6.b.

CONTROL ROD DRIVE HOUSING SUPPORT

LIMITING CONDITION FOR OPERATION

3.1.3.8 The control rod drive housing support shall be in place.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the control rod drive housing support not in place, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.8 The control rod drive housing support shall be verified to be in place by a visual inspection prior to startup any time it has been disassembled or when maintenance has been performed in the control rod drive housing support area.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

ROD WORTH MINIMIZER

LIMITING CONDITION FOR OPERATION

3.1.4.1 The rod worth minimizer (RWM) shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2[*][#], when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER, the minimum allowable low power setpoint.

ACTION:

a. With the RWM inoperable, verify control rod movement and compliance with the prescribed control rod pattern by a second licensed operator or other technically qualified member of the unit technical staff who is present at the reactor control console. Otherwise, control rod movement may be only by actuating the manual scram or placing the reactor mode switch in the Shutdown position.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.4.1 The RWM shall be demonstrated OPERABLE:

a. In OPERATIONAL CONDITION 2 within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical, and in OPERATIONAL CONDITION 1 within 8 hours prior to RWM automatic initiation when reducing THERMAL POWER, by verifying proper indication of the selection error of at least one out-of-sequence control rod.

b. In OPERATIONAL CONDITION 2 within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical, by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

c. In OPERATIONAL CONDITION 1 within one hour after RWM automatic initiation when reducing THERMAL POWER, by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

d. By verifying that the control rod patterns and sequence input to the RWM computer are correctly loaded following any loading of the program into the computer.

---

[*] Entry into OPERATIONAL CONDITION 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

[#] See Special Test Exception 3.10.2.

ROD SEQUENCE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.4.2 The rod sequence control system (RSCS) shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*#, when THERMAL POWER is less than or equal to 20% RATED THERMAL POWER, the minimum allowable low power setpoint.

ACTION:

a. With the RSCS inoperable, control rod movement shall not be permitted, except by a scram.

b. With an inoperable control rod(s), OPERABLE control rod movement may continue by bypassing the inoperable control rod(s) in the RSCS provided that:

   1. The position and bypassing of inoperable control rods is verified by a second licensed operator or other technically qualified member of the unit technical staff, and

   2. There are not more than 3 inoperable control rods in any RSCS group.

SURVEILLANCE REQUIREMENTS

4.1.4.2 The RSCS shall be demonstrated OPERABLE by:

a. Performance of a system diagnostic function:

   1. Within 8 hours prior to each reactor startup, and

   2. Prior to movement of a control rod after rod inhibit mode automatic initiation when reducing THERMAL POWER.

b. Attempting to select and move an inhibited control rod:

   1. After withdrawal of the first insequence control rod for each reactor startup, and

   2. Within one hour after rod inhibit mode automatic initiation when reducing THERMAL POWER.

---

*See Special Test Exception 3.10.2

#Entry into OPERATIONAL CONDITION 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RSCS prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

3.1.4.3 Both rod block monitor (RBM) channels shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER.

ACTION:

a. With one RBM channel inoperable:

   1. Verify that the reactor is not operating on a LIMITING CONTROL ROD PATTERN, and

   2. Restore the inoperable RBM channel to OPERABLE status within 24 hours.

   Otherwise, place the inoperable rod block monitor channel in the tripped condition within the next hour.

b. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within one hour.

SURVEILLANCE REQUIREMENTS

4.1.4.3 Each of the above required RBM channels shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies and for the OPERATIONAL CONDITIONS specified in Table 4.3.6-1.

b. CHANNEL FUNCTIONAL TEST prior to control rod withdrawal when the reactor is operating on a LIMITING CONTROL ROD PATTERN.

50

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system consists of two redundant subsystems and shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 5*

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

   1. With one system subsystem inoperable, restore the subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

   2. With both system subsystems inoperable, restore at least one subsystem to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5*:

   1. With one system subsystem inoperable, restore subsystem to OPERABLE status within 30 days or insert all insertable control rods within the next hour.

   2. With both standby liquid control system subsystems inoperable, insert all insertable control rods within one hour.

SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that;

   1. The temperature of the sodium pentaborate solution in the storage tank is greater than or equal to 70°F.

   2. The available volume of sodium pentaborate solution is within the limits of Figure 3.1.5-1.

   3. The heat tracing circuit is OPERABLE by determining the temperature of the pump suction piping to be greater than or equal to 70°F.

---

*With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

51

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by:

   1. Verifying the continuity of the explosive charge.
   2. Determining that the available weight of sodium pentaborate is greater than or equal to 5,760 lbs and the concentration of boron in solution is within the limits of Figure 3.1.5-1 by chemical analysis.*
   3. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. Demonstrating that, when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm, per pump, at a pressure of greater than or equal to 1255 psig is met.

d. At least once per 18 months during shutdown by:

   1. Initiating one of the standby liquid control system subsystem, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel and verifying that the relief valve does not actuate. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of that batch successfully fired. Both injection subsystems shall be tested in 36 months.

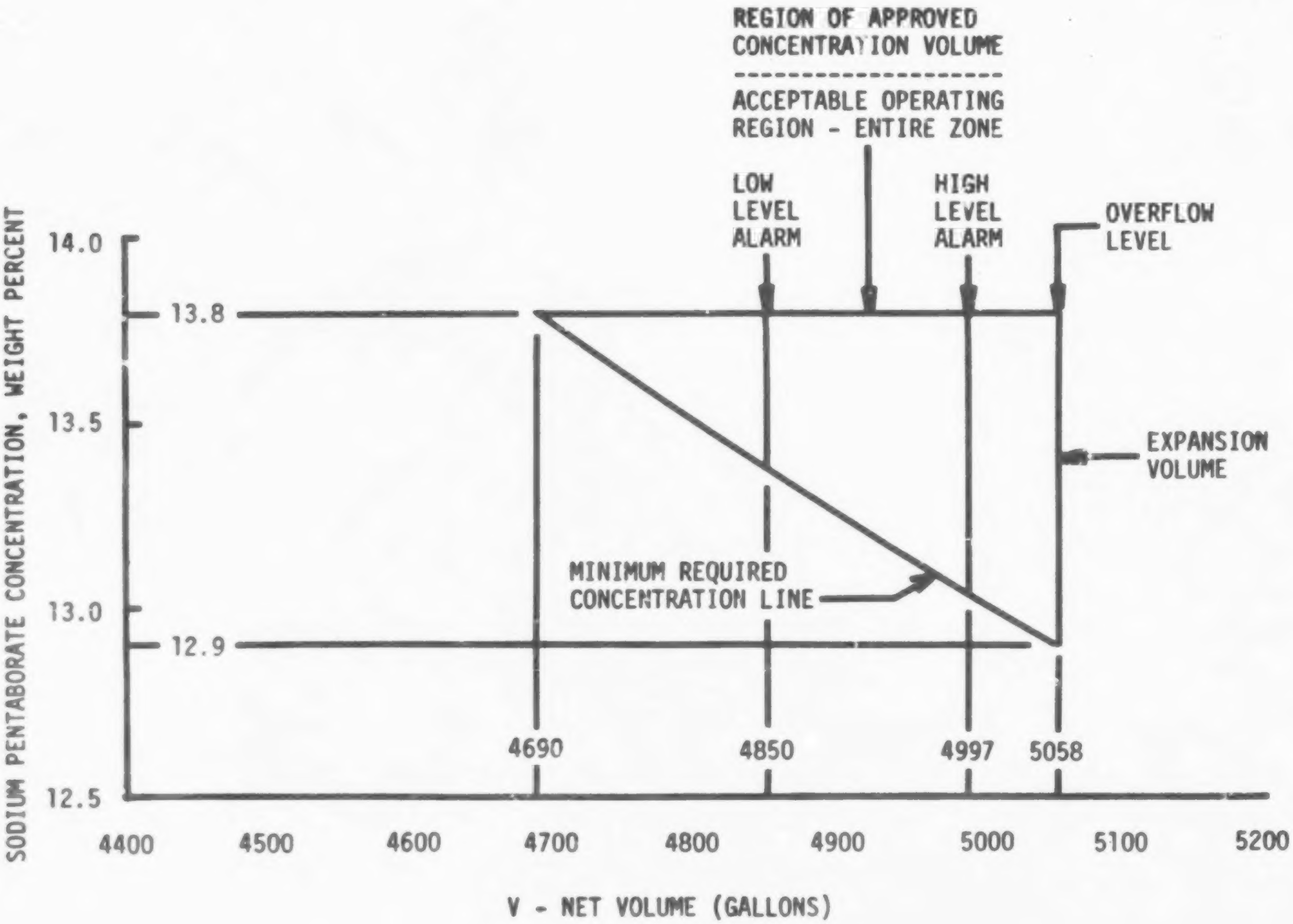
   2. **Demonstrating that all heat traced piping between the storage tank and the injection pumps is unblocked and then draining and flushing the piping with demineralized water.

   3. Demonstrating that the storage tank heaters are OPERABLE by verifying the expected temperature rise of the sodium pentaborate solution in the storage tank after the heaters are energized.

---

*This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below 70°F.

**This test shall also be performed whenever both heat tracing circuits have been found to be inoperable and may be performed by any series of sequential, overlapping or total flow path steps such that the entire flow path is included.



SODIUM PENTABORATE SOLUTION
VOLUME/CONCENTRATION REQUIREMENTS

Figure 3.1.5-1

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, and 3.2.1-5.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits of Figure 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, or 3.2.1-5, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits determined from Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4 and 3.2.1-5:

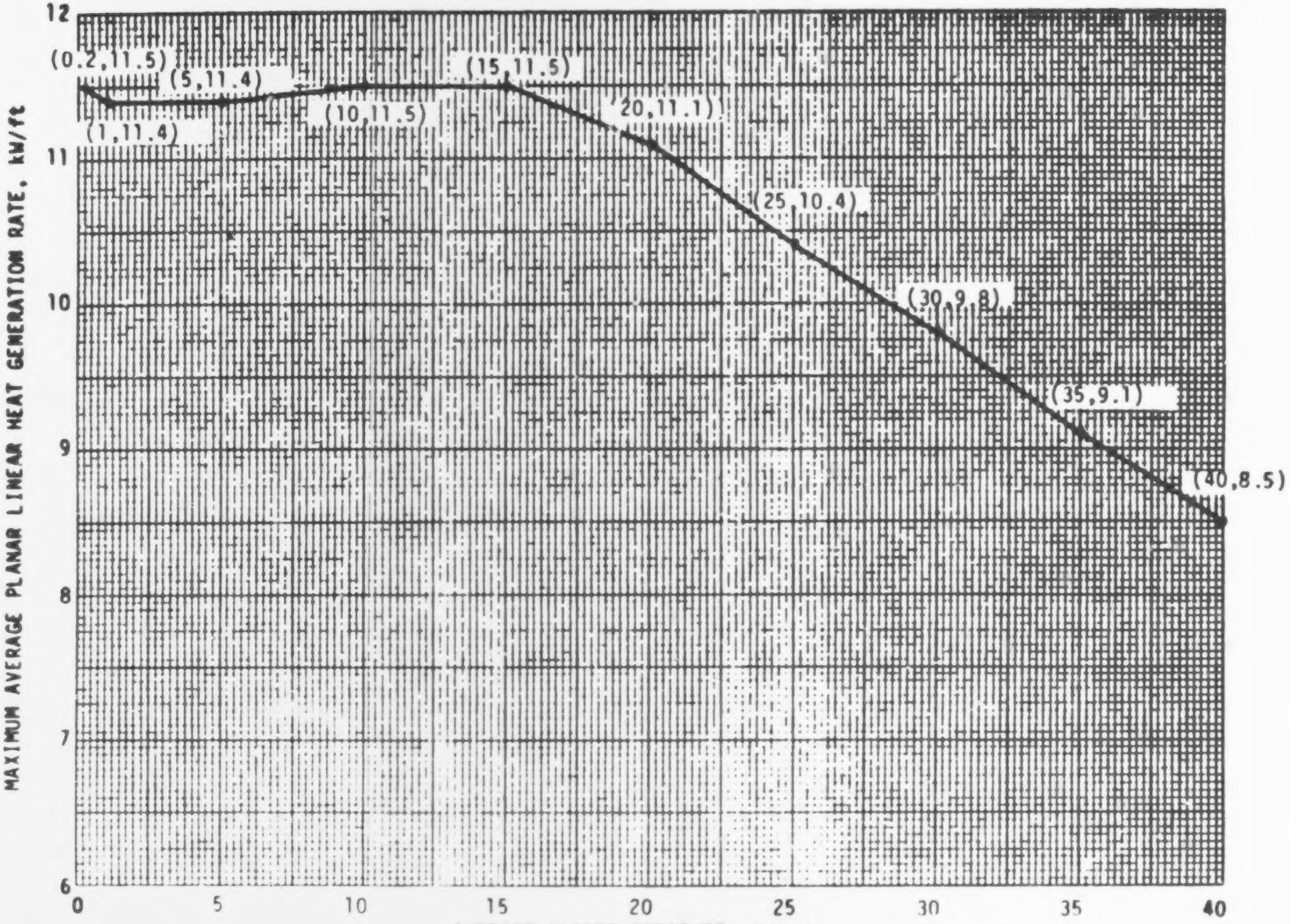
a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

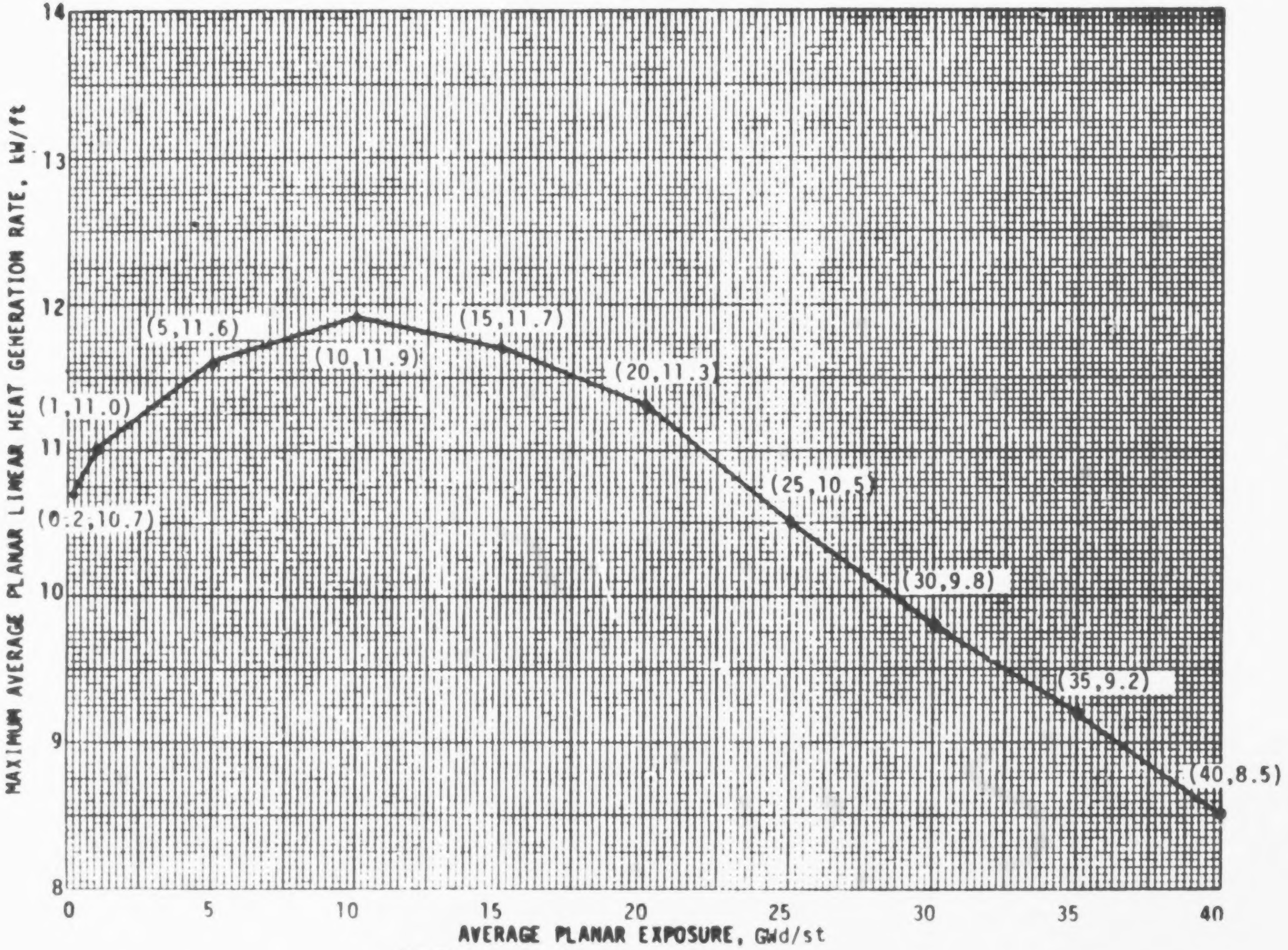
c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

d. The provisions of Specification 4.0.4 are not applicable.

54

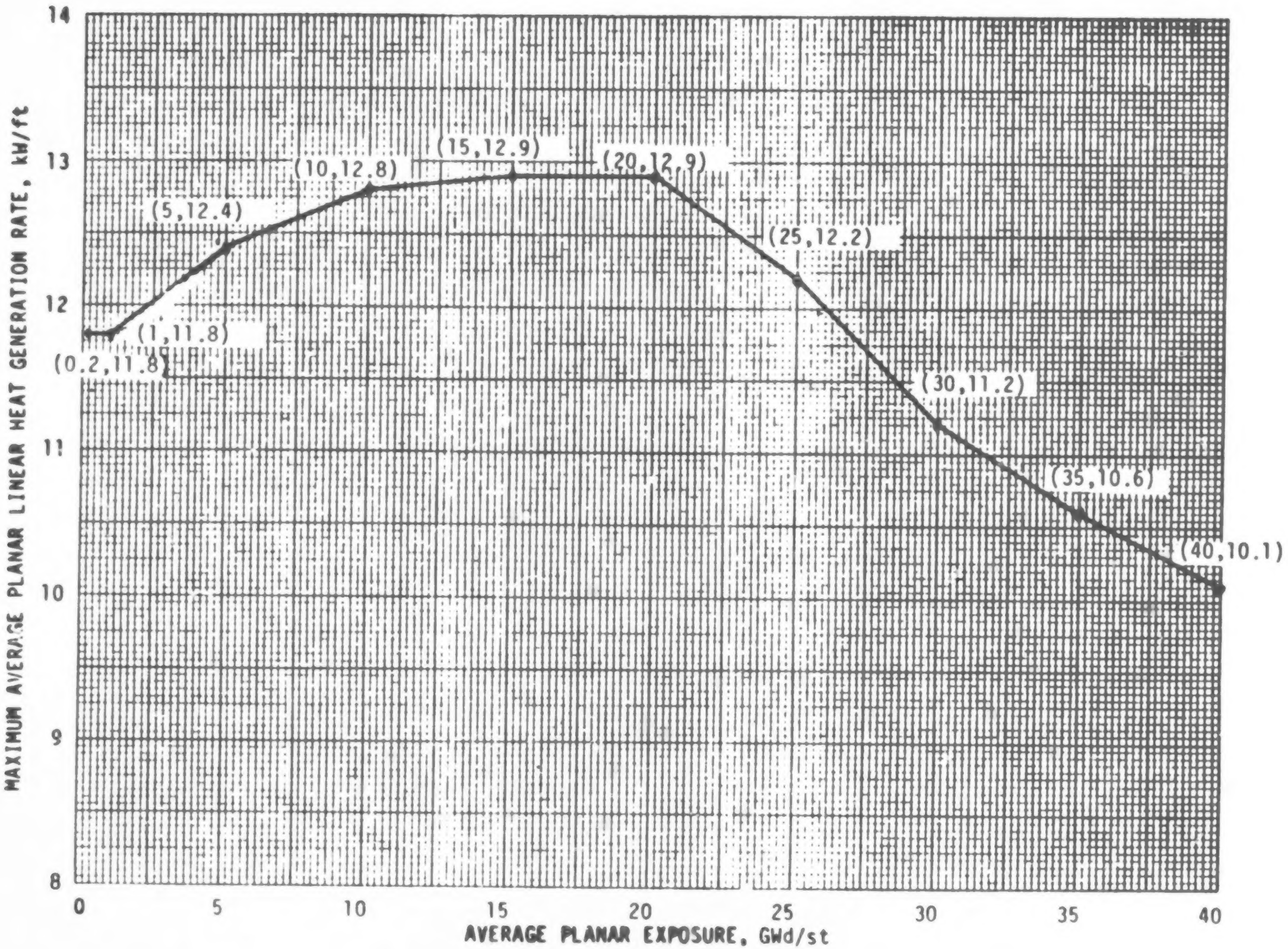


MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
INITIAL CORE FUEL TYPE P8CIB071
Figure 3.2.1-1

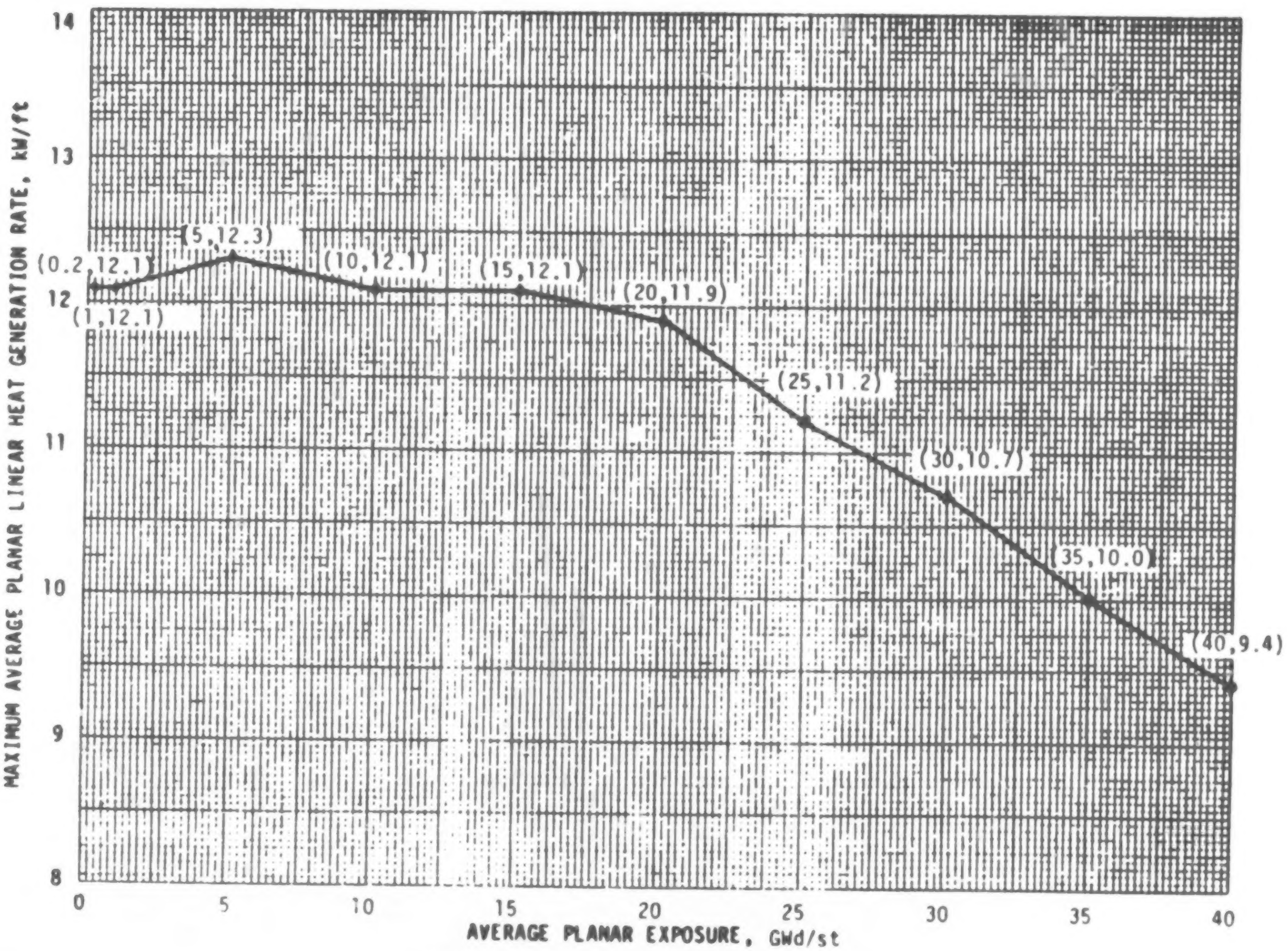


MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
INITIAL CORE FUEL TYPE P8CIB094
Figure 3.2.1-2

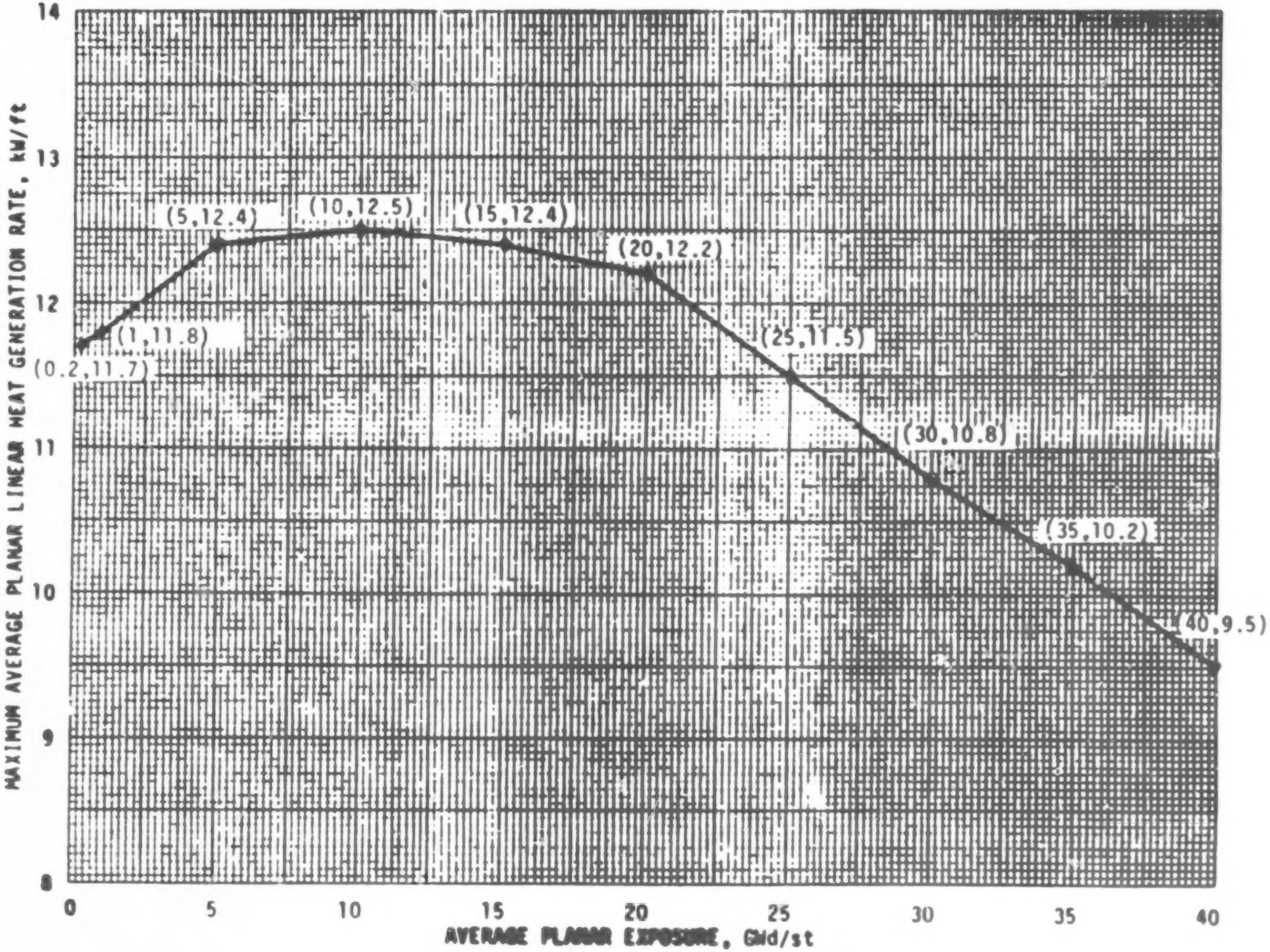
56



MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
INITIAL CORE FUEL TYPE P8CIB163
Figure 3.2.1-3



MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
INITIAL CORE FUEL TYPE P8CIB248
Figure 3.2.1-4



MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
INITIAL CORE FUEL TYPE P8CIB278
Figure 3.2.1-5

POWER DISTRIBUTION LIMITS

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow biased simulated thermal power-upscale scram trip setpoint (S) and flow biased neutron flux-upscale control rod block trip setpoint (S_{RB}) shall be established according to the following relationships:

TRIP SETPOINT	ALLOWABLE VALUE
$S \le (0.66W + 51\%)T$	$S \le (0.66W + 54\%)T$
$S_{RB} \le (0.66W + 42\%)T$	$S_{RB} \le (0.66W + 45\%)T$

where: S and S_{RB} are in percent of RATED THERMAL POWER,
W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 100 million lbs/hr,
T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER (FRTP) divided by the CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD). T is applied only if less than or equal to 1.0.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the APRM flow biased simulated thermal power-upscale scram trip setpoint and/or the flow biased neutron flux-upscale control rod block trip setpoint less conservative than the value shown in the Allowable Value column for S or S_{RB}, as above determined, initiate corrective action within 15 minutes and adjust S and/or S_{RB} to be consistent with the Trip Setpoint values* within 6 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The FRTP and the CMFLPD shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-upscale scram and flow biased neutron flux-upscale control rod block trip setpoints verified to be within the above limits or adjusted, as required:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with CMFLPD greater than or equal to FRTP.

d. The provisions of Specification 4.0.4 are not applicable.

---

*With CMFLPD greater than the FRTP up to 90% of RATED THERMAL POWER, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that the APRM readings are greater than or equal to 100% times CMFLPD provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the sum of the MCPR limit shown in Figure 3.2.3-1 plus the feedwater heating capacity adjustment given in Table 3.2.3-1 times the K_f shown in Figure 3.2.3-2, with:

$$\tau = \frac{(\tau_{ave} - \tau_B)}{\tau_A - \tau_B}$$

where:

τ_A = 0.86 seconds, control rod average scram insertion time limit to notch 39 per Specification 3.1.3.3,

$$\tau_B = 0.688 + 1.65\left[\frac{N_1}{\sum_{i=1}^{n} N_i}\right]^{\frac{1}{2}}(0.052),$$

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i},$$

n = number of surveillance tests performed to date in cycle,

N_i = number of active control rods measured in the i^{th} surveillance test,

τ_i = average scram time to notch 39 of all rods measured in the i^{th} surveillance test, and

N_1 = total number of active rods measured in Specification 4.1.3.2.a.

APPLICABILITY:

OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

ACTION:

With MCPR less than the applicable MCPR limit shown in Figures 3.2.3-1 and 3.2.3-2 plus the feedwater heating capacity adjustment given in Table 3.2.3-1, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3 MCPR, with:

a. $\tau = 1.0$ prior to performance of the initial scram time measurements for the cycle in accordance with Specification 4.1.3.2, or

b. τ as defined in Specification 3.2.3 used to determine the limit within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2,

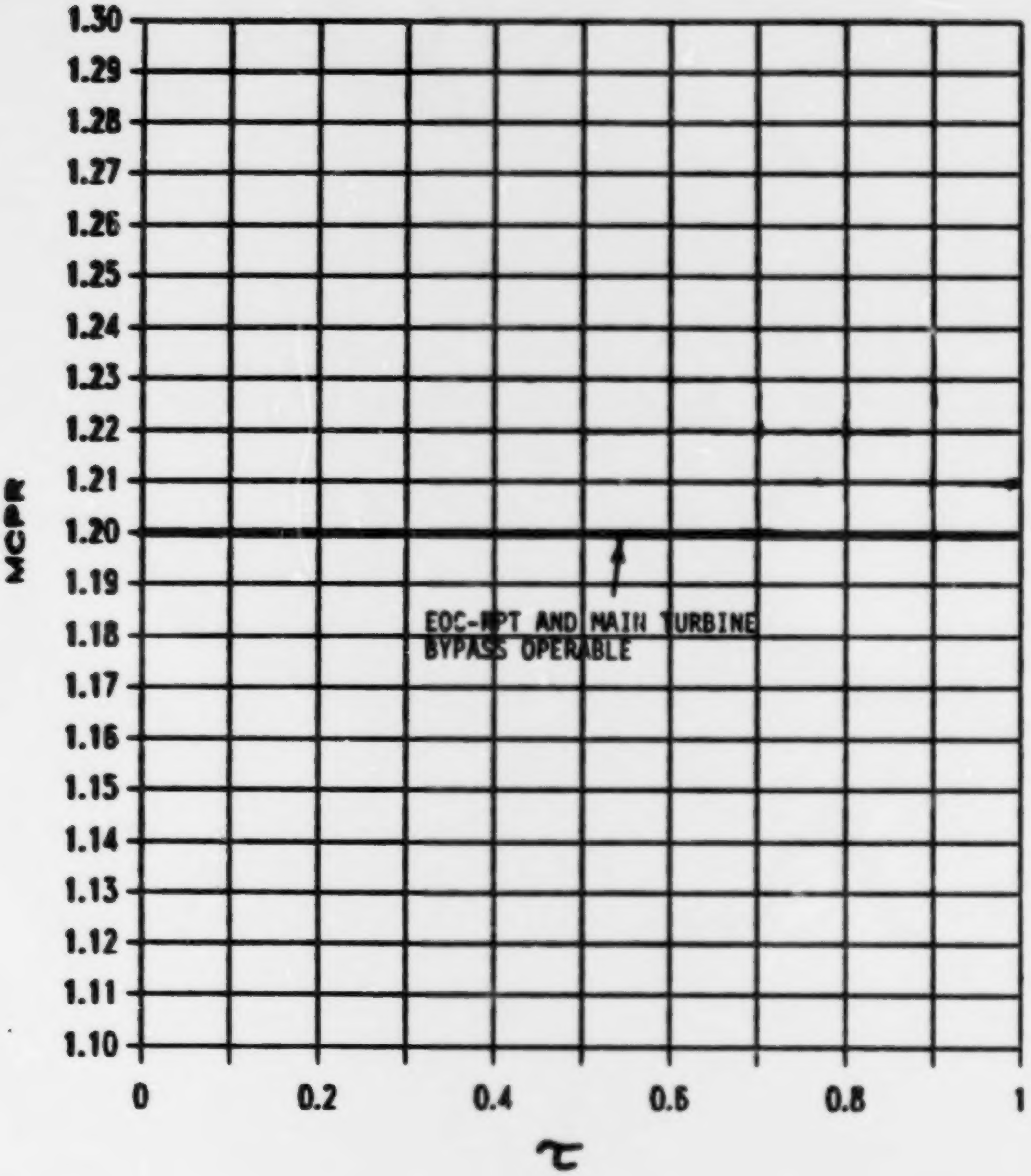
shall be determined to be equal to or greater than the applicable MCPR limit determined from Figures 3.2.3-1 and 3.2.3-2:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

d. The provisions of Specification 4.0.4 are not applicable.

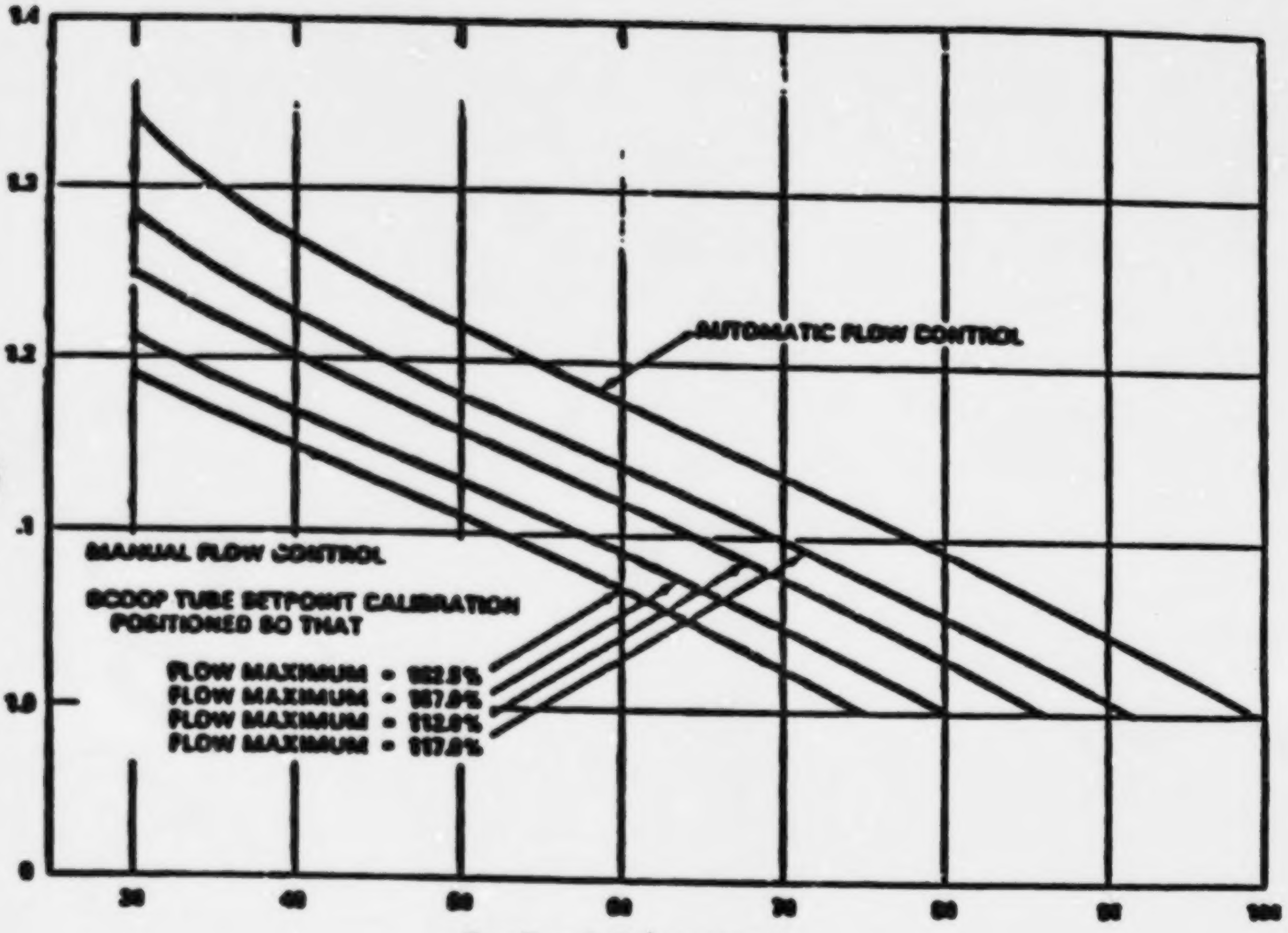


MINIMUM CRITICAL POWER RATIO (MCPR)

VERSUS τ AT RATED FLOW

Figure 3.2.3-1

63



K_f FACTOR

Figure 3.2.3-2

64

Table 3.2.3-1

MCPR FEEDWATER HEATING CAPACITY ADJUSTMENT

Rated Power Feedwater Temperature Capacity	MCPR Adjustment
≥ 400°F	0.0
≥ 320°F	0.03
≥ 250°F	0.06

65

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed 13.4 kw/ft.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the LHGR of any fuel rod exceeding the limit, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 LHGR's shall be determined to be equal to or less than the limit:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR.

d. The provisions of Specification 4.0.4 are not applicable.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within one hour. The provisions of Specification 3.0.4 are not applicable.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip system.

---

*An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.

**If more channels are inoperable in one trip system than in the other, place the trip system with more inoperable channels in the tripped condition, except when this would cause the Trip Function to occur.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors(b)			
a. Neutron Flux - High	2	3	1
	3, 4	2	2
	5(c)	3(d)	3
b. Inoperative	2	3	1
	3, 4	2	2
	5	3(d)	3
2. Average Power Range Monitor(e):			
a. Neutron Flux - Upscale, Setdown	2	2	1
	3, 4	2	2
	5(c)	2(d)	3
b. Flow Biased Simulated Thermal Power - Upscale	1	2	4
c. Fixed Neutron Flux - Upscale	1	2	4
d. Inoperative	1, 2	2	1
	3, 4	2	2
	5(c)	2(d)	3
e. Downscale	1(g)	2	4
3. Reactor Vessel Steam Dome Pressure - High	1, 2(f)	2	1
4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	1
5. Main Steam Line Isolation Valve - Closure	1(g)	4	4

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
6. Main Steam Line Radiation - High, High	1, $2^{(f)}$	2	5
7. Drywell Pressure - High	1, $2^{(h)}$	2	1
8. Scram Discharge Volume Water Level - High			
a. Float Switch	1, 2 $5^{(i)}$	2 2	1 3
b. Level Transmitter/Trip Unit	1, 2 $5^{(i)}$	2 2	1 3
9. Turbine Stop Valve - Closure	$1^{(j)}$	$4^{(k)}$	6
10. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	$1^{(j)}$	$2^{(k)}$	6
11. Reactor Mode Switch Shutdown Position	1, 2 3, 4 5	2 2 2	1 7 3
12. Manual Scram	1, 2 3, 4 5	2 2 2	1 8 9

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1 - Be in at least HOT SHUTDOWN within 12 hours.

ACTION 2 - Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.

ACTION 3 - Suspend all operations involving CORE ALTERATIONS* and insert all insertable control rods within one hour.

ACTION 4 - Be in at least STARTUP within 6 hours.

ACTION 5 - Be in STARTUP with the main steam line isolation valves closed within 6 hours or in at least HOT SHUTDOWN within 12 hours.

ACTION 6 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to less than the automatic bypass setpoint within 2 hours.

ACTION 7 - Verify all insertable control rods to be inserted within one hour.

ACTION 8 - Lock the reactor mode switch in the Shutdown position within one hour.

ACTION 9 - Suspend all operations involving CORE ALTERATIONS*, and insert all insertable control rods and lock the reactor mode switch in the SHUTDOWN position within one hour.

---

*Except replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) This function shall be automatically bypassed when the reactor mode switch is in the Run position.

(c) Unless adequate shutdown margin has been demonstrated per Specification 3.1.1, the "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn*.

(d) The non-coincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE Channels Per the Trip System are 4 APRMS, 6 IRMS and 2 SRMS.

(e) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

(f) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(g) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(h) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(j) This function shall be automatically bypassed when turbine first stage pressure is $\leq$ 153.3 psig** equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER. To allow for instrument accuracy, calibration, and drift, a setpoint of $\leq$ 132.4 psig** is used.

(k) Also actuates the EOC-RPT system.

---

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

**Initial setpoint. Final setpoint to be determined during the startup test program.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - Upscale, Setdown	NA
b. Flow Biased Simulated Thermal Power - Upscale	≤ 0.09**
c. Fixed Neutron Flux - Upscale	≤ 0.09
d. Inoperative	NA
e. Downscale	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 0.55
4. Reactor Vessel Water Level - Low, Level 3	≤ 1.05
5. Main Steam Line Isolation Valve - Closure	≤ 0.06
6. Main Steam Line Radiation - High, High	NA
7. Drywell Pressure - High	NA
8. Scram Discharge Volume Water Level - High	NA
a. Float Switch	NA
b. Level Transmitter/Trip Unit	NA
9. Turbine Stop Valve - Closure	≤ 0.06
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≤ 0.08#
11. Reactor Mode Switch Shutdown Position	NA
12. Manual Scram	NA

---

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.

**Not including simulated thermal power time constant, 6 ± 0.6 seconds.

#Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U[b],S	S/U[c], W	R	2
	S	W	R	3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor[f]:				
a. Neutron Flux - Upscale, Setdown	S/U[b],S	S/U[c], W	SA	2
	S	W	SA	3, 4, 5
b. Flow Biased Simulated Thermal Power - Upscale	S,D[g]	S/U[c], W	W[d][e],SA,R[h]	1
c. Fixed Neutron Flux - Upscale	S	S/U[c], W	W[d], SA	1
d. Inoperative	NA	W	NA	1, 2, 3, 4, 5
e. Downscale	S	W	SA	1
3. Reactor Vessel Steam Dome Pressure - High	S	M	R	1, 2
4. Reactor Vessel Water Level - Low, Level 3	S	M	R	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	M	R	1
6. Main Steam Line Radiation - High, High	S	M	R	1, 2[i]
7. Drywell Pressure - High	S	M	R	1, 2

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High				
a. Float Switch	NA	M	R	1, 2, 5$^{(j)}$
b. Level Transmitter/Trip Unit	S	M	R	1, 2, 5$^{(j)}$
9. Turbine Stop Valve - Closure	NA	M	R	1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA	M	R	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	M	NA	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least ½ decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least ½ decades during each controlled shutdown, if not performed within the previous 7 days.

(c) Within 24 hours prior to startup, if not performed within the previous 7 days.

(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.

(g) Verify measured core flow (total core flow) to be greater than or equal to established core flow at the existing recirculation loop flow (APRM % flow).

(h) This calibration shall consist of verifying the 6 ± 0.6 second simulated thermal power time constant.

(i) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(j) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The isolation actuation instrumentation channels shown in Table 3.3.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.2-2 and with ISOLATION SYSTEM RESPONSE TIME as shown in Table 3.3.2-3.

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

a. With an isolation actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within one hour. The provisions of Specification 3.0.4 are not applicable.

c. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.2-1.

---

*An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken.

**The trip system need not be placed in the tripped condition if this would cause the Trip Function to occur. When a trip system can be placed in the tripped condition without causing the Trip Function to occur, place the trip system with the most inoperable channels in the tripped condition; if both systems have the same number of inoperable channels, place either trip system in the tripped condition.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each isolation actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.2.1-1.

4.3.2.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.2.3 The ISOLATION SYSTEM RESPONSE TIME of each isolation trip function shown in Table 3.3.2-3 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months, where N is the total number of redundant channels in a specific isolation trip system.

76

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE ACTUATION GROUPS OPERATED BY SIGNAL(d)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low Low, Level 2	1, 2, 8, 9, 12, 13, 14, 15, 17, 18	2	1, 2, 3	20
2) Low Low Low, Level 1	10, 11, 15, 16	2	1, 2, 3	20
b. Drywell Pressure - High	1, 8, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18	2	1, 2, 3	20
c. Reactor Building Exhaust Radiation - High	1, 8, 9, 12 13, 14, 15, 17, 18	3	1, 2, 3	28
d. Manual Initiation	1, 8, 9, 10 11, 12, 13, 14, 15, 16, 17, 18	1	1, 2, 3	24
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low, Level 2	19(c)	2	1, 2, 3 and *	26
b. Drywell Pressure - High	19(c)	2	1, 2, 3	26
c. Refueling Floor Exhaust Radiation - High	19(c)	3	1, 2, 3 and *	29
d. Reactor Building Exhaust Radiation - High	19(c)	3	1, 2, 3 and *	28
e. Manual Initiation	19(c)	1	1, 2, 3 and *	26

77

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION		VALVE ACTUATION GROUPS OPERATED BY SIGNAL$^{(d)}$	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM $^{(a)}$	APPLICABLE OPERATIONAL CONDITION	ACTION
3.	**MAIN STEAM LINE ISOLATION**				
a.	Reactor Vessel Water Level - Low Low Low, Level 1	1	2	1, 2, 3	21
b.	Main Steam Line Radiation - High, High	1, 2$^{(b)}$	2	1, 2, 3	21
c.	Main Steam Line Pressure - Low	1	2	1	22
d.	Main Steam Line Flow - High	1	2/line	1, 2, 3	20
e.	Condenser Vacuum - Low	1	2	1, 2**, 3**	21
f.	Main Steam Line Tunnel Temperature - High	1	2/line	1, 2, 3	21
g.	Manual Initiation	1, 2, 17	2	1, 2, 3	25
4.	**REACTOR WATER CLEANUP SYSTEM ISOLATION**				
a.	RWCU Δ Flow - High	7	1/Valve$^{(e)}$	1, 2, 3	23
b.	RWCU Δ Flow - High, Timer	7	1/Valve$^{(e)}$	1, 2, 3	23
c.	RWCU Area Temperature - High	7	6/Valve$^{(e)}$	1, 2, 3	23
d.	RWCU Area Ventilation Δ Temperature-High	7	6/Valve$^{(e)}$	1, 2, 3	23
e.	SLCS Initiation	7$^{(f)}$	1/Valve$^{(e)}$	1, 2, 5#	23
f.	Reactor Vessel Water Level - Low Low, Level 2	7	2/Valve$^{(e)}$	1, 2, 3	23
g.	Manual Initiation	7	1/Valve$^{(e)}$	1, 2, 3	25

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE ACTUATION GROUPS OPERATED BY SIGNAL[(d)]	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]	APPLICABLE OPERATIONAL CONDITION	ACTION
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Δ Pressure (Flow) - High	6	1/Valve[(e)]	1, 2, 3	23
b. RCIC Steam Line Δ Pressure (Flow) - High, Timer	6	1/Valve[(e)]	1, 2, 3	23
c. RCIC Steam Supply Pressure - Low	6	2/Valve[(e)]	1, 2, 3	23
d. RCIC Turbine Exhaust Diaphragm Pressure - High	6	2/Valve[(e)]	1, 2, 3	23
e. RCIC Pump Room Temperature - High	6	1/Valve[(e)]	1, 2, 3	23
f. RCIC Pump Room Ventilation Ducts Δ Temperature - High	6	1/Valve[(e)]	1, 2, 3	23
g. RCIC Pipe Routing Area Temperature - High	6	1/Valve[(e)]	1, 2, 3	23
h. RCIC Torus Compartment Temperature-High	6	3/Valve[(e)]	1, 2, 3	23
i. Drywell Pressure - High[(g)]	6	2/Valve[(e)]	1, 2, 3	23
j. Manual Initiation	6[(h)]	1/RCIC System	1, 2, 3	25

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION		VALVE ACTUATION GROUPS OPERATED BY SIGNAL(d)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
6.	HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION				
a.	HPCI Steam Line Δ Pressure (Flow) - High	5	1/Valve(e)	1, 2, 3	23
b.	HPCI Steam Line Δ Pressure (Flow) - High, Timer	5	1/Valve(e)	1, 2, 3	23
c.	HPCI Steam Supply Pressure-Low	5	2/Valve(e)	1, 2, 3	23
d.	HPCI Turbine Exhaust Diaphragm Pressure - High	5	2/Valve(e)	1, 2, 3	23
e.	HPCI Pump Room Temperature - High	5	1/Valve(e)	1, 2, 3	23
f.	HPCI Pump Room Ventilation Ducts Δ Temperature - High	5	1/Valve(e)	1, 2, 3	23
g.	HPCI Pipe Routing Area Temperature - High	5	1/Valve(e)	1, 2, 3	23
h.	HPCI Torus Compartment Temperature - High	5	3/Valve(e)	1, 2, 3	23
i.	Drywell Pressure - High(g)	5	2/Valve(e)	1, 2, 3	23
j.	Manual Initiation	5(i)	1/HPCI system	1, 2, 3	25

80

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE ACTUATION GROUPS OPERATED BY SIGNAL(d)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
7. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	3	2/Valve(e)	1, 2, 3	27
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	3	2/Valve(e)	1, 2, 3	27
c. Manual Initiation	3	1/Valve(e)	1, 2, 3	25

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 21 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 22 - Be in at least STARTUP within 6 hours.

ACTION 23 - Close the affected system isolation valves within one hour and declare the affected system inoperable.

ACTION 24 - Restore the manual initiation function to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 25 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable.

ACTION 26 - Establish SECONDARY CONTAINMENT INTEGRITY with the Filtration, Recirculation and Ventilation System (FRVS) operating within one hour.

ACTION 27 - Lock the affected system isolation valves closed within one hour and declare the affected system inoperable.

ACTION 28 - Place the inoperable channel in the tripped condition or close the affected system isolation valves within one hour and declare the affected system inoperable.

ACTION 29 - Place the inoperable channel in the tripped condition or establish SECONDARY CONTAINMENT INTEGRITY with the Filtration, Recirculation, and Ventilation System (FRVS) operating within one hour.

NOTES

* When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

** When any turbine stop valve is greater than 90% open and/or when the key-locked bypass switch is in the Norm position.

# Refer to Specification 3.1.5 for applicability.

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

(b) Also trips and isolates the mechanical vacuum pumps.

(c) Also starts the Filtration, Recirculation and Ventilation System (FRVS).

(d) Refer to Table 3.3.2-1 table notation for the listing of which valves in an actuation group are closed by a particular isolation signal. Refer to Tables 3.6.3-1 and 3.6.5.2-1 for the listings of all valves within an actuation group.

(e) Sensors arranged per valve group, not per trip system.

(f) Closes only RWCU system isolation valve(s) HV-F001 and HV-F004.

(g) Requires system steam supply pressure-low coincident with drywell pressure-high to close turbine exhaust vacuum breaker valves.

(h) Manual isolation closes HV-F008 only, and only following manual or automatic initiation of the RCIC system.

(i) Manual isolation closes HV-F003 and HV-F042 only, and only following manual or automatic initiation of the HPCI system.

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TABLE NOTATION

This table notation identifies which valves, in an actuation group, are closed by a particular trip signal. If all valves in the group are closed by the trip signal, only the valve group number will be listed. If only certain valves in the group are closed by the trip signal, the valve group number will be listed followed by, in parentheses, a listing of which valves are closed by the trip signal.

TRIP FUNCTION	VALVES CLOSED BY SIGNAL
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level -	
1) Low Low, Level 2	1 (HV-5834A, HV-5835A, HV-5836A, HV-5837A), 2, 8, 9, 12, 13, 14, 15 (HV-5154, HV-5155), 17, 18
2) Low Low Low, Level 1	10, 11, 15(HV-5126 A&B, HV-5152 A&B, HV-5147, HV-5148, HV-5162), 16
b. Drywell Pressure - High	1 (HV-5834A, HV-5835A, HV-5836A, HV-5837A), 8, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18
c. Reactor Building Exhaust Radiation - High	1 (HV-5834A, HV-5835A, HV-5836A, HV-5837A), 8, 9, 12, 13, 14, 15, 17 (HV-5161), 18
d. Manual Initiation	1 (HV-5834A, HV-5835A, HV-5836A, HV-5837A), 8, 9, 10, 11, 12, 13, 14, 15, 16, 17 (HV-5161), 18
2. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level - Low Low, Level 2	19
b. Drywell Pressure - High	19
c. Refueling Floor Exhaust Radiation - High	19
d. Reactor Building Exhaust Radiation - High	19
e. Manual Initiation	19

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TABLE NOTATION

TRIP FUNCTION	VALVES CLOSED BY SIGNAL
3. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level - Low Low Low, Level 1	1 (HV-F022A, B, C & D, HV-F028A, B, C & D, HV-F067A, B, C & D, HV-F016, HV-F019)
b. Main Steam Line Radiation - High, High	1 (as above), 2
c. Main Steam Line Presure - Low	1 (as above)
d. Main Steam Line Flow - High	1 (as above)
e. Condenser Vacuum - Low	1 (as above)
f. Main Steam Line Tunnel Temperature - High	1 (as above)
g. Manual Initiation	1 (as above), 2, 17 (SV-J004A-1, 2, 3, 4 & 5)
4. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. RWCU Δ Flow - High	7
b. RWCU Δ Flow - High, Timer	7
c. RWCU Area Temperature - High	7

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TABLE NOTATION

TRIP FUNCTION		VALVES CLOSED BY SIGNAL
d.	RWCU Area Ventilation Δ Temperature - High	7
e.	SLCS Initiation	7
f.	Reactor Vessel Water Level - Low Low, Level 2	7
g.	Manual Initiation	7
5.	REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a.	RCIC Steam Line Δ Pressure (Flow) - High	6 (HV-F007, HV-F076, HV-F008)
b.	RCIC Steam Line Δ Pressure (Flow) - High, Timer	6 (HV-F007, HV-F076, HV-F008)
c.	RCIC Steam Supply Pressure - Low	6
d.	RCIC Turbine Exhaust Diaphragm Pressure - High	6 (HV-F007, HV-F076, HV-F008)
e.	RCIC Pump Room Temperature - High	6 (HV-F007, HV-F076, HV-F008)
f.	RCIC Pump Room Ventilation Ducts Δ Temperature - High	6 (HV-F007, HV-F076, HV-F008)
g.	RCIC Pipe Routing Area Temperature - High	6 (HV-F007, HV-F076, HV-F008)
h.	RCIC Torus Compartment Temperature - High	6 (HV-F007, HV-F076, HV-F008)

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TABLE NOTATION

TRIP FUNCTION	VALVES CLOSED BY SIGNAL
i. Drywell Pressure - High	6 (HV-F062, HV-F084)
j. Manual Initiation	6 (HV-F008)
6. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION	
a. HPCI Steam Line Δ Pressure (Flow) - High	5 (HV-F002, HV-F100, HV-F003, HV-F042)
b. HPCI Steam Line Δ Pressure (Flow) - High, Timer	5 (HV-F002, HV-F100, HV-F003, HV-F042)
c. HPCI Steam Supply Pressure - Low	5
d. HPCI Turbine Exhaust Diaphragm Pressure - High	5 (HV-F002, HV-F100, HV-F003, HV-F042)
e. HPCI Pump Room Temperature - High	5 (HV-F002, HV-F100, HV-F003, HV-F042)
f. HPCI Pump Room Ventilation Ducts Δ Temperature - High	5 (HV-F002, HV-F100, HV-F003, HV-F042)
g. HPCI Pipe Routing Area Temperature - High	5 (HV-F002, HV-F100, HV-F003, HV-F042)
h. HPCI Torus Compartment Temperature - High	5 (HV-F002, HV-F100, HV-F003, HV-F042)
i. Drywell Pressure - High	5 (HV-F075, HV-F079)
j. Manual Initiation	5 (HV-F003, HV-F042)

BLANK PAGE

BLANK PAGE

BLANK PAGE

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TABLE NOTATION

TRIP FUNCTION	VALVES CLOSED BY SIGNAL
7. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION	
a. Reactor Vessel Water Level Low, Level 3	3 (HV-F008, HV-F009, HV-F015A & B, HV-F022, HV-F023)
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	3 (HV-F008, HV-F009, HV-F015A & B, HV-F022, HV-F023)
c. Manual Initiation	3 (HV-F008, HV-F009, HV-F015A & B, HV-F022, HV-F023)

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level		
1) Low Low, Level 2	≥ -38.0 inches*	≥ -45.0 inches
2) Low Low Low, Level 1	≥ -129.0 inches*	≥ -136.0 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Reactor Building Exhaust Radiation - High	≤ 1×10^{-3} μCi/cc**	≤ 1.2×10^{-3} μCi/cc**
d. Manual Initiation	NA	NA
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -38.0 inches*	≥ -45.0 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Refueling Floor Exhaust Radiation - High	≤ 2×10^{-3} μCi/cc**	≤ 2.4×10^{-3} μCi/cc**
d. Reactor Building Exhaust Radiation - High	≤ 1×10^{-3} μCi/cc**	≤ 1.2×10^{-3} μCi/cc**
e. Manual Initiation	NA	NA
3. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -129.0 inches*	≥ -136.0 inches
b. Main Steam Line Radiation - High, High	≤ 3.0 X full power background	≤ 3.6 X full power background
c. Main Steam Line Pressure - Low	≥ 756.0 psig	≥ 736.0 psig
d. Main Steam Line Flow - High	≤ 108.7 psid	≤ 111.7 psid

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
MAIN STEAM LINE ISOLATION (Continued)		
e. Condenser Vacuum - Low	≥ 8.5 inches Hg vacuum	≥ 7.6 inches Hg vacuum
f. Main Steam Line Tunnel Temperature - High	≤ 160°F	≤ 172°F
g. Manual Initiation	NA	NA
4. **REACTOR WATER CLEANUP SYSTEM ISOLATION**		
a. RWCU Δ Flow - High	≤ 56.3 gpm	≤ 61.3 gpm
b. RWCU Δ Flow - High, Timer	45.0 seconds ≤ t ≤ 47.0 seconds	45.0 seconds ≤ t ≤ 47.0 seconds
c. RWCU Area Temperature - High	≤ 160°F, 140°F or 135°F***	≤ 172°F, 152°F or 147°F***
d. RWCU/Area Ventilation Δ Temperature - High	≤ 60°F	≤ 70°F
e. SLCS Initiation	NA	NA
f. Reactor Vessel Water Level - Low Low, Level 2	≥ -38.0 inches*	≥ -45.0 inches
g. Manual Initiation	NA	NA
5. **REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION**		
a. RCIC Steam Line Δ Pressure (Flow) - High	≤ 609.6" H_2O**	≤ 622.1" H_2O**
b. RCIC Steam Line Δ Pressure (Flow) - High, Timer	3.0 seconds ≤ t ≤ 13.0 seconds	3.0 seconds ≤ t ≤ 13.0 seconds
c. RCIC Steam Supply Pressure - Low	≥ 64.5 psig	≥ 56.5 psig
d. RCIC Turbine Exhaust Diaphragm Pressure - High	≤ 10.0 psig	≤ 20.0 psig

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

	TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
	REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION (Continued)		
e.	RCIC Pump Room Temperature - High	≤ 160°F	≤ 172°F
f.	RCIC Pump Room Ventilation Duct Δ Temperature - High	≤ 70°F	≤ 80°F
g.	RCIC Pipe Routing Area Temperature - High	≤ 160°F$^{\#}$	≤ 172°F$^{\#}$
h.	RCIC Torus Compartment Temperature - High	≤ 128°F$^{\#}$	≤ 140°F$^{\#}$
i.	Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
j.	Manual Initiation	NA	NA
6.	**HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION**		
a.	HPCI Steam Line Δ Pressure (Flow) - High	≤ 1025.0 inches H_2O**	≤ 1044.5 inches H_2O**
b.	HPCI Steam Line Δ Pressure (Flow) - High, Timer	3.0 seconds ≤ t ≤ 13.0 seconds	3.0 seconds ≤ t ≤ 13.0 seconds
c.	HPCI Steam Supply Pressure - Low	≥ 100.0 psig	≥ 90.0 psig
d.	HPCI Turbine Exhaust Diaphragm Pressure - High	≤ 10.0 psig	≤ 20.0 psig
e.	HPCI Pump Room Temperature - High	≤ 160°F	≤ 172°F
f.	HPCI Pump Room Ventilation Ducts Δ Temperature - High	≤ 70°F	≤ 80°F
g.	HPCI Pipe Routing Area Temperature - High	≤ 160°F$^{\#\#}$	≤ 172°F$^{\#\#}$
h.	HPCI Torus Compartment Temperature - High	≤ 128°F$^{\#\#}$	≤ 140°F$^{\#\#}$
i.	Drywell Pressure High	≤ 1.68 psig	≤ 1.88 psig
j.	Manual Initiation	NA	NA

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
7. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		
a. Reactor Vessel Water Level - Low, Level 3	≥ 12.5 inches*	≥ 11.0 inches
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	≤ 82.0 psig	≤ 102.0 psig
c. Manual Initiation	NA	NA

---

*See Bases Figure B 3/4 3-1.

**Initial setpoint. Final setpoint to be determined during startup test program.

***These setpoints are as follows:
160°F - RWCU pipe chase room 4402
140°F - RWCU pump room and heat exchanger rooms
135°F - RWCU pipe chase room 4505

#30 minute time delay.

##15 minute time delay.

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. **PRIMARY CONTAINMENT ISOLATION**	
a. Reactor Vessel Water Level	
1) Low Low, Level 2	NA
2) Low Low Low, Level 1	NA
b. Drywell Pressure - High	NA
c. Reactor Building Exhaust Radiation - High	NA
d. Manual Initiation	NA
2. **SECONDARY CONTAINMENT ISOLATION**	
a. Reactor Vessel Water Level-Low Low, Level 2	NA
b. Drywell Pressure - High	NA
c. Refueling Floor Exhaust Radiation - High$^{(b)}$	$\leq$ 4.0
d. Reactor Building Exhaust Radiation - High$^{(b)}$	$\leq$ 4.0
e. Manual Initiation	NA
3. **MAIN STEAM LINE ISOLATION**	
a. Reactor Vessel Water Level - Low Low Low, Level 1	$\leq$ 1.0*/$\leq$ 13$^{(a)}$**
b. Main Steam Line Radiation - High, High$^{(a)(b)}$	$\leq$ 1.0*/$\leq$ 13$^{(a)}$**
c. Main Steam Line Pressure - Low	$\leq$ 1.0*/$\leq$ 13$^{(a)}$**
d. Main Steam Line Flow-High	$\leq$ 0.5*/$\leq$ 13$^{(a)}$**
e. Condenser Vacuum - Low	NA
f. Main Steam Line Tunnel Temperature - High	NA
g. Manual Initiation	NA
4. **REACTOR WATER CLEANUP SYSTEM ISOLATION**	
a. RWCU Δ Flow - High	NA
b. RWCU Δ Flow - High, Timer	NA
c. RWCU Area Temperature - High	NA
d. RWCU Area Ventilation Δ Temperature - High	NA
e. SLCS Initiation	NA
f. Reactor Vessel Water Level - Low Low, Level 2	NA
g. Manual Initiation	NA
5. **REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION**	
a. RCIC Steam Line Δ Pressure (Flow) - High	NA
b. RCIC Steam Line Δ Pressure (Flow) - High, Timer	NA
c. RCIC Steam Supply Pressure - Low	NA
d. RCIC Turbine Exhaust Diaphragm Pressure - High	NA

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION		RESPONSE TIME (Seconds)#
REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
e.	RCIC Pump Room Temperature - High	NA
f.	RCIC Pump Room Ventilation Ducts Δ Temperature - High	NA
g.	RCIC Pipe Routing Area Temperature - High	NA
h.	RCIC Torus Compartment Temperature - High	NA
i.	Drywell Pressure - High	NA
j.	Manual Initiation	NA
6.	HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION	
a.	HPCI Steam Line Δ Pressure (Flow) - High	NA
b.	HPCI Steam Line Δ Pressure (Flow) - High, Timer	NA
c.	HPCI Steam Supply Pressure - Low	NA
d.	HPCI Turbine Exhaust Diaphragm Pressure - High	NA
e.	HPCI Pump Room Temperature - High	NA
f.	HPCI Pump Room Ventilation Ducts Δ Temperature - High	NA
g.	HPCI Pipe Routing Area Temperature - High	NA
h.	HPCI Torus Compartment Temperature - High	NA
i.	Drywell Pressure - High	NA
j.	Manual Initiation	NA
7.	RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION	
a.	Reactor Vessel Water Level - Low, Level 3	NA
b.	Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA
c.	Manual Initiation	NA

(a) Isolation system instrumentation response time specified includes diesel generator starting and sequence loading delays.

(b) Radiation detectors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel.

*Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed for MSIVs.

**Isolation system instrumentation response time for associated valves except MSIVs.

#Isolation system instrumentation response time specified for the Trip Function actuating each valve group shall be added to isolation time shown in Table 3.6.3-1 and 3.6.5.2-1 for valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level -				
1) Low Low, Level 2	S	M	R	1, 2, 3
2) Low Low Low, Level 1	S	M	R	1, 2, 3
b. Drywell Pressure - High	S	M	R	1, 2, 3
c. Reactor Building Exhaust Radiation - High	S	M	R	1, 2, 3
d. Manual Initiation	NA	M(a)	NA	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low, Level 2	S	M	R	1, 2, 3 and *
b. Drywell Pressure - High	S	M	R	1, 2, 3
c. Refueling Floor Exhaust Radiation - High	S	M	R	1, 2, 3 and *
d. Reactor Building Exhaust Radiation - High	S	M	R	1, 2, 3 and *
e. Manual Initiation	NA	M(a)	NA	1, 2, 3 and *
3. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R	1, 2, 3
b. Main Steam Line Radiation - High, High	S	M	R	1, 2, 3
c. Main Steam Line Pressure - Low	S	M	R	1
d. Main Steam Line Flow - High	S	M	R	1, 2, 3

94

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION		CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
MAIN STEAM LINE ISOLATION (Continued)					
e.	Condenser Vacuum - Low	S	M	R	1, 2**, 3**
f.	Main Steam Line Tunnel Temperature - High	NA	M	R	1, 2, 3
g.	Manual Initiation	NA	$M^{(a)}$	NA	1, 2, 3
4.	REACTOR WATER CLEANUP SYSTEM ISOLATION				
a.	RWCU Δ Flow - High	S	M	R	1, 2, 3
b.	RWCU Δ Flow - High, Timer	NA	M	R	1, 2, 3
c.	RWCU Area Temperature - High	NA	M	R	1, 2, 3
d.	RWCU Area Ventilation Δ Temperature - High	NA	M	R	1, 2, 3
e.	SLCS Initiation	NA	$M^{(b)}$	NA	1, 2, $5^{\#}$
f.	Reactor Vessel Water Level - Low Low, Level 2	S	M	R	1, 2, 3
g.	Manual Initiation	NA	$M^{(a)}$	NA	1, 2, 3
5.	REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a.	RCIC Steam Line Δ Pressure (Flow) - High	NA	M	R	1, 2, 3
b.	RCIC Steam Line Δ Pressure (Flow) - High, Timer	NA	M	R	1, 2, 3
c.	RCIC Steam Supply Pressure - Low	NA	M	R	1, 2, 3
d.	RCIC Turbine Exhaust Diaphragm Pressure - High	NA	M	R	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION (Continued)				
e. RCIC Pump Room Temperature - High	NA	M	R	1, 2, 3
f. RCIC Pump Room Ventilation Ducts Δ Temperature - High	NA	M	R	1, 2, 3
g. RCIC Pipe Routing Area Temperature - High	NA	M	R	1, 2, 3
h. RCIC Torus Compartment Temperature -High	NA	M	R	1, 2, 3
i. Drywell Pressure - High	S	M	R	1, 2, 3
j. Manual Initiation	NA	R	NA	1, 2, 3
6. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION				
a. HPCI Steam Line Δ Pressure (Flow) - High	NA	M	R	1, 2, 3
b. HPCI Steam Line Δ Pressure (Flow) - High, Timer	NA	M	R	1, 2, 3
c. HPCI Steam Supply Pressure - Low	NA	M	R	1, 2, 3
d. HPCI Turbine Exhaust Diaphragm Pressure - High	NA	M	R	1, 2, 3
e. HPCI Pump Room Temperature - High	NA	M	R	1, 2, 3
f. HPCI Pump Room Ventilation Ducts Δ Temperature - High	NA	M	R	1, 2, 3
g. HPCI Pipe Routing Area Temperature - High	NA	M	R	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION (Continued)				
h. HPCI Torus Compartment Temperature - High	NA	M	R	1, 2, 3
i. Drywell Pressure - High	NA	M	R	1, 2, 3
j. Manual Initiation	NA	R	NA	1, 2, 3
7. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	S	M	R	1, 2, 3
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA	M	R	1, 2, 3
c. Manual Initiation	NA	M(a)	NA	1, 2, 3

---

* When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

** When any turbine stop valve is greater than 90% open and/or when the key-locked bypass switch is in the Norm position.

# Refer to Specification 3.1.5 for applicability.

(a) Manual initiation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days as part of circuitry required to be tested for automatic system isolation.

(b) Each train or logic channel shall be tested at least every other 31 days.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3 The emergency core cooling system (ECCS) actuation instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2 and with EMERGENCY CORE COOLING SYSTEM RESPONSE TIME as shown in Table 3.3.3-3.

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more ECCS actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each ECCS actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.3.1-1.

4.3.3.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.3.3 The ECCS RESPONSE TIME of each ECCS trip function shown in Table 3.3.3-3 shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ECCS trip system.

98

TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION$^{(a)}$	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. CORE SPRAY SYSTEM			
a. Reactor Vessel Water Level - Low Low Low, Level 1	$2^{(b)(e)}$	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	$2^{(b)(e)}$	1, 2, 3	30
c. Reactor Vessel Pressure - Low (Permissive)	4/division$^{(f)}$	1, 2, 3	31
		4*, 5*	32
d. Core Spray Pump Discharge Flow - Low (Bypass)	1/subsystem	1, 2, 3, 4*, 5*	37
e. Core Spray Pump Start Time Delay - Normal Power	1/subsystem	1, 2, 3, 4*, 5*	31
f. Core Spray Pump Start Time Delay - Emergency Power	1/subsystem	1, 2, 3, 4*, 5*	31
g. Manual Initiation	1/division$^{(b)(g)}$	1, 2, 3, 4*, 5*	33
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2/valve	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2/valve	1, 2, 3	30
c. Reactor Vessel Pressure - Low (Permissive)	1/valve	1, 2, 3	31
		4*, 5*	32
d. LPCI Pump Discharge Flow - Low (Bypass)	1/pump	1, 2, 3, 4*, 5*	37
e. LPCI Pump Start Time Delay - Normal Power	1/pump$^{(i)}$	1, 2, 3, 4*, 5*	31
f. Manual Initiation	1/subsystem	1, 2, 3, 4*, 5*	33
3. HIGH PRESSURE COOLANT INJECTION SYSTEM#			
a. Reactor Vessel Water Level - Low Low Level 2	4	1, 2, 3	34
b. Drywell Pressure - High	4	1, 2, 3	34
c. Condensate Storage Tank Level - Low	$2^{(c)}$	1, 2, 3	35
d. Suppression Pool Water Level - High	$2^{(c)}$	1, 2, 3	35
e. Reactor Vessel Water Level - High, Level 8	$4^{(d)}$	1, 2, 3	31
f. HPCI Pump Discharge Flow - Low (Bypass)	1	1, 2, 3	37
g. Manual Initiation	1/system	1, 2, 3	33
4. AUTOMATIC DEPRESSURIZATION SYSTEM##			
a. Reactor Vessel Water Level - Low Low Low, Level 1	4	1, 2, 3	30
b. Drywell Pressure - High	4	1, 2, 3	30
c. ADS Timer	2	1, 2, 3	31
d. Core Spray Pump Discharge Pressure - High (Permissive)	1/pump	1, 2, 3	31

TABLE 3.3.3-1 (Cont'd)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION$^{(a)}$	APPLICABLE OPERATIONAL CONDITIONS	ACTION
4. AUTOMATIC DEPRESSURIZATION SYSTEM##			
e. RHR LPCI Mode Pump Discharge Pressure - High (Permissive)	2/pump	1, 2, 3	31
f. Reactor Vessel Water Level - Low, Level 3 (Permissive)	2	1, 2, 3	31
g. ADS Drywell Pressure Bypass Timer	4	1, 2, 3	31
h. ADS Manual Inhibit Switch	2	1, 2, 3	31
i. Manual Initiation	4	1, 2, 3	33

TRIP FUNCTION	TOTAL NO OF CHANNELS$^{(h)}$	CHANNELS TO TRIP$^{(h)}$	MINIMUM CHANNELS OPERABLE$^{(h)}$	APPLICABLE OPERATIONAL CONDITIONS	ACTION
5. LOSS OF POWER					
1. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	4/bus	2/bus	3/bus	1, 2, 3, 4**, 5**	36
2. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	2/source/ bus	2/source/ bus	2/source/ bus	1, 2, 3, 4**, 5**	36

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) Also actuates the associated emergency diesel generators.

(c) One trip system. Provides signal to HPCI pump suction valve only.

(d) Provides a signal to trip HPCI pump turbine only.

(e) In divisions 1 and 2, the two sensors are associated with each pump and valve combination. In divisions 3 and 4, the two sensors are associated with each pump only.

(f) Division 1 and 2 only.

(g) In divisions 1 and 2, manual initiation is associated with each pump and valve combination; in divisions 3 and 4, manual initiation is associated with each pump only.

(h) Each voltage detector is a channel.

(i) Start time delay is applicable to LPCI Pump C and D only.

* When the system is required to be OPERABLE per Specification 3.5.2.

** Required when ESF equipment is required to be OPERABLE.

Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 200 psig.

Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. With one channel inoperable, place the inoperable channel in the tripped condition within one hour* or declare the associated system inoperable.

b. With more than one channel inoperable, declare the associated system inoperable.

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, declare the associated ECCS inoperable.

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

ACTION 33 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 8 hours or declare the associated ECCS inoperable.

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. For one channel inoperable, place the inoperable channel in the tripped condition within 1 hour* or declare the HPCI system inoperable.

b. With more than one channel inoperable, declare the HPCI system inoperable.

ACTION 35 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour* or declare the HPCI system inoperable.

ACTION 36 - With the number of OPERABLE channels one less than the Total Number of Channels, place the inoperable channel in the tripped condition within 1 hour;* operation may then continue until performance of the next required CHANNEL FUNCTIONAL TEST.

ACITON 37 - With the number of OPERABLE channels less than required by the Minimum OPERABLE channels per Trip Function requirement, open the minimum flow bypass valve within one hour. Restore the inoperable channel to OPERABLE status within 7 days or declare the associated ECCS inoperable.

---

*The provisions of Specification 3.0.4 are not applicable.

101

TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION		TRIP SETPOINT	ALLOWABLE VALUE
1.	**CORE SPRAY SYSTEM**		
a.	Reactor Vessel Water Level - Low Low Low, Level 1	≥-129 inches*	≥-136 inches
b.	Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c.	Reactor Vessel Pressure - Low	461 psig	≤ 481 psig and ≥ 441 psig
d.	Core Spray Pump Discharge Flow - Low (Bypass)	≥ 775 gpm	≥ 650 gpm
e.	Core Spray Pump Start Time Delay - Normal Power	10 seconds	≥ 9 seconds and ≤ 11 seconds
f.	Core Spray Pump Start Time Delay - Emergency Power	6 seconds	≥ 5 seconds and ≤ 7 seconds
g.	Manual Initiation	NA	NA
2.	**LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM**		
a.	Reactor Vessel Water Level - Low Low Low, Level 1	≥-129 inches*	≥-136 inches
b.	Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c.	Reactor Vessel Pressure - Low (Permissive)	450 psig	≤ 460 psig and ≥ 440 psig
d.	LPCI Pump Discharge Flow - Low (Bypass)	≥ 1250 gpm	≥ 1100 gpm
e.	LPCI Pump Start Time Delay - Normal Power	5 seconds	≥ 4 seconds and ≤ 6 seconds
f.	Manual Initiation	NA	NA
3.	**HIGH PRESSURE COOLANT INJECTION SYSTEM**		
a.	Reactor Vessel Water Level - (Low Low, Level 2)	≥-38 inches*	≥-45 inches
b.	Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c.	Condensate Storage Tank Level - Low	≥ 22,558 gallons	≥ 19,174 gallons
d.	Suppression Pool Water Level - High	≤ 78.5 inches	≤ 80.3 inches
e.	Reactor Vessel Water Level - High, Level 8	≤ 54 inches	≤ 61 inches
f.	HPCI Pump Discharge Flow - Low (Bypass)	≥ 550 gpm	≥ 500 gpm
g.	Manual Initiation	NA	NA

102

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
4. AUTOMATIC DEPRESSURIZATION SYSTEM		
a. Reactor Water Level - Low Low Low, Level 1	≥-129 inches*	≥-136 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. ADS Timer	≤ 105 seconds	≤ 117 seconds
d. Core Spray Pump Discharge Pressure - High	145 psig	≤ 155 psig ≥ 125 psig
e. RHR LPCI Mode Pump Discharge Pressure-High	125 psig	≤ 135 psig ≥ 115 psig
f. Reactor Vessel Water Level-Low, Level 3	≥ 12.5 inches	≥ 11.0 inches
g. ADS Drywell Pressure Bypass Timer	≤ 5.0 minutes	≤ 5.5 minutes
h. ADS Manual Inhibit Switch	NA	NA
i. Manual Initiation	NA	NA
5. LOSS OF POWER		
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	a. 4.16 kv Basis - 2975 ± 30 volts	2975 ± 63 volts
	b. 120 v Basis - 85 ± 0.85 volts	85 ± 1.8 volts
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)**	a. 4.16 kv Basis - ≥ 3857 volts	≥ 3857 volts
	b. 120 v Basis - ≥ 110.2 volts	≥ 109.0 volts
	c. 20 sec @ 109.0 volts	20 + 15, - 5 sec @ 109.0 volts

---

* See Bases Figure B 3/4 3-1.

** This is an inverse time delay voltage relay. The voltages shown are the maximum that will not result in a trip. Some voltage conditions will result in decreased trip times.

TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

ECCS	RESPONSE TIME (Seconds)
1. CORE SPRAY SYSTEM	≤ 27
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM	≤ 40
3. AUTOMATIC DEPRESSURIZATION SYSTEM	NA
4. HIGH PRESSURE COOLANT INJECTION SYSTEM	≤ 27
5. LOSS OF POWER	NA

104

TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. CORE SPRAY SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	M	R	1, 2, 3
c. Reactor Vessel Pressure - Low	S	M	R	1, 2, 3, 4*, 5*
d. Core Spray Pump Discharge Flow - Low (Bypass)	S	M	R	1, 2, 3, 4*, 5*
e. Core Spray Pump Start Time Delay - Normal Power	NA	M	R	1, 2, 3, 4*, 5*
f. Core Spray Pump Start Time Delay - Emergency Power	NA	M	R	1, 2, 3, 4*, 5*
g. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	M	R	1, 2, 3
c. Reactor Vessel Pressure - Low (Permissive)	S	M	R	1, 2, 3, 4*, 5*
d. LPCI Pump Discharge FLow - Low (Bypass)	S	M	R	1, 2, 3, 4*, 5*
e. LPCI Pump Start Time Delay - Normal Power	NA	M	R	1, 2, 3, 4*, 5*
f. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
3. HIGH PRESSURE COOLANT INJECTION SYSTEM#				
a. Reactor Vessel Water Level - Low Low, Level 2	S	M	R	1, 2, 3
b. Drywell Pressure - High	S	M	R	1, 2, 3
c. Condensate Storage Tank Level - Low	S	M	R	1, 2, 3
d. Suppression Pool Water Level - High	S	M	R	1, 2, 3
e. Reactor Vessel Water Level - High, Level 8	S	M	R	1, 2, 3
f. HPCI Pump Discharge FLow - Low (Bypass)	S	M	R	1, 2, 3
g. Manual Initiation	NA	R	NA	1, 2, 3

105

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
4. AUTOMATIC DEPRESSURIZATION SYSTEM##				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R	1, 2, 3
b. Drywell Pressure - High	S	M	R	1, 2, 3
c. ADS Timer	NA	M	Q	1, 2, 3
d. Core Spray Pump Discharge Pressure - High	S	M	R	1, 2, 3
e. RHR LPCI Mode Pump Discharge Pressure - High	S	M	R	1, 2, 3
f. Reactor Vessel Water Level - Low, Level 3	S	M	R	1, 2, 3
g. ADS Drywell Pressure Bypass Timer	NA	M	Q	1, 2, 3
h. ADS Manual Inhibit Switch	NA	R	NA	1, 2, 3
i. Manual Initiation	NA	R	NA	1, 2, 3
5. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	NA	NA	R	1, 2, 3, 4**, 5**
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	S	M	R	1, 2, 3, 4**, 5**

---

* When the system is required to be OPERABLE per Specification 3.5.2.

** Required OPERABLE when ESF equipment is required to be OPERABLE.

# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 200 psig.

## Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.1 The anticipated transient without scram recirculation pump trip (ATWS-RPT) system instrumentation channels shown in Table 3.3.4.1-1 shall be OPERABLE with their trip setpoints set consistent with values shown in the Trip Setpoint column of Table 3.3.4.1-2.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

a. With an ATWS recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.1-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within one hour.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, and:

    1. If the inoperable channels consist of one reactor vessel water level channel and one reactor vessel pressure channel, place both inoperable channels in the tripped condition within one hour, or if this action will initiate a pump trip, declare the trip system inoperable.

    2. If the inoperable channels include two reactor vessel water level channels or two reactor vessel pressure channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.4.1.1. Each ATWS recirculation pump trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.1-1.

4.3.4.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

TABLE 3.3.4.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]
1. Reactor Vessel Water Level - Low Low, Level 2	2
2. Reactor Vessel Pressure - High	2

(a) One channel may be placed in an inoperable status for up to 2 hours for required surveillance provided the other channel is OPERABLE.

108

TABLE 3.3.4.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel, Water Level - Low Low, Level 2	≥ -38 inches*	≥ -45 inches
2. Reactor Vessel Pressure - High	≤ 1071 psig	≤ 1086 psig

---

*See Bases Figure B3/4 3-1.

TABLE 4.3.4.1-1

ATWS RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Vessel Water Level - Low Low, Level 2	S	M	R
2. Reactor Vessel Pressure - High	S	M	R

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.4.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.4.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER.

ACTION:

a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within one hour.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and:

    1. If the inoperable channels consist of one turbine control valve channel and one turbine stop valve channel, place both inoperable channels in the tripped condition within one hour.

    2. If the inoperable channels include two turbine control valve channels or two turbine stop valve channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or reduce THERMAL POWER to less than 30% of RATED THERMAL POWER within the next 6 hours.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or reduce THERMAL POWER to less than 30% of RATED THERMAL POWER within the next 6 hours.

///

SURVEILLANCE REQUIREMENTS

4.3.4.2.1 Each end-of-cycle recirculation pump trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.2.1-1.

4.3.4.2.2. LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.4.2.3 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME of each trip function shown in Table 3.3.4.2-3 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least the logic of one type of channel input, turbine control valve fast closure or turbine stop valve closure, such that both types of channel inputs are tested at least once per 36 months.

4.3.4.2.4 The time interval necessary for breaker arc suppression from energization of the recirculation pump circuit breaker trip coil shall be measured at least once per 60 months.

112

TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM(a)
1. Turbine Stop Valve - Closure	2(b)
2. Turbine Control Valve-Fast Closure	2(b)

(a) A trip system may be placed in an inoperable status for up to 2 hours for required surveillance provided that the other trip system is OPERABLE.

(b) This function shall be automatically bypassed when turbine first stage pressure is $\leq$ 153.3 psig* equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER. To allow for instrument accuracy, calibration and drift, a setpoint of $\leq$ 132.4 psig* is used.

---

*Initial Setpoint. Final setpoint to be determined during the startup test program.

TABLE 3.3.4.2-2

END-OF-CYCLE RECIRCULATION PUMP TRIP SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Turbine Stop Valve-Closure	≤ 5% closed	≤ 7% closed
2. Turbine Control Valve-Fast Closure	≥ 530 psig	≥ 465 psig

TABLE 3.3.4.2-3

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Milleseconds)
1. Turbine Stop Valve-Closure	≤ 175
2. Turbine Control Valve-Fast Closure	≤ 175

115

TABLE 4.3.4.2.1-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Turbine Stop Valve-Closure	M	R
2. Turbine Control Valve-Fast Closure	M	R

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5 The reactor core isolation cooling (RCIC) system actuation instrumentation channels shown in Table 3.3.5-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.5-2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

a. With a RCIC system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.5-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more RCIC system actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.5-1.

SURVEILLANCE REQUIREMENTS

4.3.5.1 Each RCIC system actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.5.1-1.

4.3.5.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

TABLE 3.3.5-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION(a)	ACTION
a. Reactor Vessel Water Level - Low Low, Level 2	4(b)	50
b. Reactor Vessel Water Level - High, Level 8	4(b)	50
c. Condensate Storage Tank Water Level - Low(e)	2(c)	51
d. Manual Initiation	1(d)	52

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided all other channels monitoring that parameter are OPERABLE.

(b) One trip system with one-out-of-two twice logic.

(c) One trip system with one-out-of-two logic.

(d) One trip system with one channel.

(e) Initiates RCIC suction switchover from the condensate storage tank to the torus.

118

TABLE 3.3.5-1 (Continued)

REACTOR CORE ISOLATION COOLING SYSTEM

ACTUATION INSTRUMENTATION

ACTION 50 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. With one channel inoperable, place the inoperable channel in the tripped condition within 1 hour or declare the RCIC system inoperable.

b. With more than one channel inoperable, declare the RCIC system inoperable.

ACTION 51 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour or declare the RCIC system inoperable.

ACTION 52 - With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 8 hours or declare the RCIC system inoperable.

TABLE 3.3.5-2

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTIONAL UNITS	TRIP SETPOINT	ALLOWABLE VALUE
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -38 inches*	≥ -45 inches
b. Reactor Vessel Water Level - High, Level 8	≤ 54 inches*	≤ 61 inches
c. Condensate Storage Tank Level - Low	≥ 22,558 gallons	≥ 19,174 gallons
d. Manual Initiation	NA	NA

---

*See Bases Figure B 3/4 3-1.

TABLE 4.3.5.1-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Water Level - Low Low, Level 2	S	M	R
b. Reactor Vessel Water Level - High, Level 8	S	M	R
c. Condensate Storage Tank Level - Low	NA	M	R
d. Manual Initiation	NA	M(a)	NA

(a) Manual initiation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days as part of circuitry required to be tested for automatic system actuation.

121

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6. The control rod block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.

TABLE 3.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. **ROD BLOCK MONITOR**(a)			
a. Upscale	2	1*	60
b. Inoperative	2	1*	60
c. Downscale	2	1*	60
2. **APRM**			
a. Flow Biased Neutron Flux - Upscale	4	1	61
b. Inoperative	4	1, 2, 5	61
c. Downscale	4	1	61
d. Neutron Flux - Upscale, Startup	4	2, 5	61
3. **SOURCE RANGE MONITORS**			
a. Detector not full in(b)	3	2	61
	2	5	61
b. Upscale(c)	3	2	61
	2	5	61
c. Inoperative(c)	3	2	61
	2	5	61
d. Downscale(d)	3	2	61
	2	5	61
4. **INTERMEDIATE RANGE MONITORS**			
a. Detector not full in	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative	6	2, 5	61
d. Downscale(e)	6	2, 5	61
5. **SCRAM DISCHARGE VOLUME**			
a. Water Level-High (Float Switch)	2	1, 2, 5**	62
6. **REACTOR COOLANT SYSTEM RECIRCULATION FLOW**			
a. Upscale	2	1	62
b. Inoperative	2	1	62
c. Comparator	2	1	62
7. **REACTOR MODE SWITCH SHUTDOWN POSITION**	2	3, 4	63

TABLE 3.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

ACTION

ACTION 60 - Declare the RBM inoperable and take the ACTION required by Specification 3.1.4.3.

ACTION 61 - With the number of OPERABLE Channels:

a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

The provisions of Specification 3.0.4 are not applicable.

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

The provisions of Specification 3.0.4 are not applicable.

ACTION 63 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, initiate a rod block.

NOTES

* With THERMAL POWER $\geq$ 30% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

a. The RBM shall be automatically bypassed when a peripheral control rod is selected.

b. This function shall be automatically bypassed if detector count rate is > 100 cps or the IRM channels are on range 3 or higher.

c. This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

d. This function shall be automatically bypassed when the IRM channels are on range 3 or higher.

e. This function shall be automatically bypassed when the IRM channels are on range 1.

124

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale	≤ 0.66 W + 40%	≤ 0.66 W + 43%
b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux - Upscale	≤ 0.66 W + 42%*	≤ 0.66 W + 45%*
b. Inoperative	NA	NA
c. Downscale	≥ 4% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale, Startup	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 1.0×10^5 cps	≤ 1.6×10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥ 3 cps**	≥ 1.8 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 108/125 divisions of full scale	≤ 110/125 divisions of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High (Float Switch)	109'1" (North Volume) 108'11.5" (South Volume)	109'3" (North Volume) 109'1.5" (South Volume)
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ 108% of rated flow	≤ 111% of rated flow
b. Inoperative	NA	NA
c. Comparator	≤ 10% flow deviation	≤ 11% flow deviation
7. REACTOR MODE SWITCH SHUTDOWN POSITION	NA	NA

---

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

**May be reduced to 0.7 cps provided the signal-to-noise ratio is ≥ 2.

TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION$^{(a)}$	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	NA	S/U$^{(b)(c)}$, M$^{(c)}$	SA	1*
b. Inoperative	NA	S/U$^{(b)(c)}$, M$^{(c)}$	NA	1*
c. Downscale	NA	S/U$^{(b)(c)}$, M$^{(c)}$	SA	1*
2. APRM				
a. Flow Biased Neutron Flux - Upscale	NA	S/U$^{(b)}$, M	SA	1
b. Inoperative	NA	S/U$^{(b)}$, M	NA	1, 2, 5
c. Downscale	NA	S/U$^{(b)}$, M	SA	1
d. Neutron Flux - Upscale, Startup	NA	S/U$^{(b)}$, M	SA	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U$^{(b)}$, W	NA	2, 5
b. Upscale	NA	S/U$^{(b)}$, W	SA	2, 5
c. Inoperative	NA	S/U$^{(b)}$, W	NA	2, 5
d. Downscale	NA	S/U$^{(b)}$, W	SA	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U$^{(b)}$, W	NA	2, 5
b. Upscale	NA	S/U$^{(b)}$, W	SA	2, 5
c. Inoperative	NA	S/U$^{(b)}$, W	NA	2, 5
d. Downscale	NA	S/U$^{(b)}$, W	SA	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High (Float Switch)	NA	M	R	1, 2, 5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	S/U$^{(b)}$, M	SA	1
b. Inoperative	NA	S/U$^{(b)}$, M	NA	1
c. Comparator	NA	S/U$^{(b)}$, M	SA	1
7. REACTOR MODE SWITCH SHUTDOWN POSITION	NA	R	NA	3, 4

126

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

b. Within 24 hours prior to startup, if not performed within the previous 7 days.

c. Includes reactor manual control multiplexing system input.

* With THERMAL POWER ≥ 30% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

127

3/4.3.7 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.1 The radiation monitoring instrumentation channels shown in Table 3.3.7.1-1 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3.7.1-1.

ACTION:

a. With a radiation monitoring instrumentation channel alarm/trip setpoint exceeding the value shown in Table 3.3.7.1-1, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION required by Table 3.3.7.1-1.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.1 Each of the above required radiation monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the conditions and at the frequencies shown in Table 4.3.7.1-1.

TABLE 3.3.7.1-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	ACTION
1. Control Room Ventilation Radiation Monitor	2/intake	1,2,3,5 and *	$\leq 2\times10^{-5}\mu C/cc$**	71
2. Area Monitors				
a. Criticality Monitors				
1) New Fuel Storage Vault	1	#	$\geq$ 5 mR/hr and $\leq$ 20 mR/hr(a)	72
2) Spent Fuel Storage Pool	1	##	$\geq$ 5 mR/hr and $\leq$ 20 mR/hr(a)	72
b. Control Room Direct Radiation Monitor	1	At all times	2.5 mR/hr(a)	72
3. Reactor Auxiliaries Cooling Radiation Monitor	1	At all times	9×10^{-5} μC/cc(a)	73
4. Safety Auxiliaries Cooling Radiation Monitor	1/loop	At all times	6×10^{-5} μC/cc(a)	73
5. Offgas Pre-treatment Radiation Monitor	1	***	(b)	74

129

TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

TABLE NOTATION

*When irradiated fuel is being handled in the secondary containment.
**Activates control room emergency filtration system.
***When the offgas treatment system is operating.
#With fuel in the new fuel storage vault.
##With fuel in the spent fuel storage pool.
(a)Alarm only.
(b)Alarm setpoint to be set in accordance with Specification 3.11.2.7.

TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

ACTION

ACTION 71 -

a. With one of the required monitors inoperable, place the inoperable channel in the tripped condition within one hour; restore the inoperable channel to OPERABLE status within 7 days, or, within the next 6 hours, initiate and maintain operation of the control room emergency filtration system in the pressurization mode of operation.

b. With both of the required monitors inoperable, initiate and maintain operation of the control room emergency filtration system in the pressurization mode of operation within one hour.

ACTION 72 - With the required monitor inoperable, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 73 - With the required monitor inoperable, obtain and analyze at least one sample of the monitored parameter at least once per 24 hours.*

ACTION 74 - With the number of channels OPERABLE less than required by Minimum Channels OPERABLE requirement, release(s) via this pathway may continue for up to 30 days provided:

a. The offgas system is not bypassed, and

b. Grab samples are taken at least once per 8 hours and analyzed within the following 4 hours;

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

---

*Radiation level readings may be taken at the Local Radiation Processor (LRP) at least once per 24 hours in lieu of obtaining and analyzing grab samples at least once per 24 hours prior to 120 days after initial fuel load.

131

BEST COPY AVAILABLE

TABLE 4.3.7.1-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Control Room Ventilation Radiation Monitor	S	M	R	1, 2, 3, 5 and *
2. Area Monitors				
a. Criticality Monitors				
1) New Fuel Storage Vault	S	M	R	#
2) Spent Fuel Storage Pool	S	M	R	##
b. Control Room Direct Radiation Monitor	S	M	R	At all times
3. Reactor Auxiliaries Cooling Radiation Monitor	S	M	R	At all times
4. Safety Auxiliaries Cooling Radiation Monitor	S	M	R	At all times
5. Offgas Pre-treatment Radiation Monitor	S	M	R	**

TABLE 4.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

#With fuel in the new fuel storage vault.
##With fuel in the spent fuel storage pool.
*When irradiated fuel is being handled in the secondary containment.
**When the offgas treatment system is operating.

SEISMIC MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.2 The seismic monitoring instrumentation shown in Table 3.3.7.2-1 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.2.1 Each of the above required seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.2-1.

4.3.7.2.2 Each of the above required seismic monitoring instruments actuated during a seismic event greater than or equal to 0.01g shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 5 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum and resultant effect upon unit features important to safety.

TABLE 3.3.7.2-1

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelerographs		
a. 500' From Reactor Building Free Field, 60' Below Grade	± 1G	1
b. Primary Containment Foundation, Room 4101	± 1G	1
c. Refueling Floor in Reactor Building	± 1G	1
d. Core Spray Piping in Drywell	± 1G	1
e. Auxiliary Building Foundation	± 1G	1
2. Triaxial Peak Accelerographs		
a. Reactor Support Lateral Truss	± 5G	1
b. Core Spray Piping in Drywell	± 5G	1
c. Service Water Pump Piping	± 5G	1
3. Triaxial Seismic Switches		
a. Primary Containment Foundation, Room 4101 (Trigger)	NA	1*
b. Primary Containment Foundation, Room 4101 (Switch)	NA	1(a)
4. Triaxial Response-Spectrum Recorders		
a. Primary Containment Foundation (north-south)	1.0 -32.0 Hz**	1
b. Primary Containment Foundation (east-west)	1.0 -32.0 Hz**	1
c. Primary Containment Foundation (vertical)	1.0 -32.0 Hz**	1

(a) With reactor control room annunciation.

*Provides trigger mechanism to activate magnetic recording tapes for the time-history accelerographs.

**Each recorder has 16 reeds responsive to 16 discrete frequencies from 1.0-32.0 Hz. Each recorder also contains 16 switches integrally related to the 16 reeds which provide independent control room indication when predetermined acceleration levels and design limits have been exceeded.

135

TABLE 4.3.7.2-1

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Triaxial Time-History Accelerographs			
a. 500' From Reactor Building Free Field, 60' Below Grade	M	SA	R
b. Primary Containment Foundation, Room 4101	M	SA	R
c. Refueling Floor in Reactor Building	M	SA	R
d. Core Spray Piping in Drywell	M	SA	R
e. Auxiliary Building Foundation	M	SA	R
2. Triaxial Peak Accelerographs			
a. Reactor Support Lateral Truss	NA	NA	R
b. Core Spray Piping in Drywell	NA	NA	R
c. Service Water Pump Piping	NA	NA	R
3. Triaxial Seismic Switches			
a. Primary Containment Foundation, Room 4101 (Trigger)	NA	SA	R
b. Primary Containment Foundation Room 4101 (Switch)	NA	SA	R
4. Triaxial Response-Spectrum Recorders			
a. Primary Containment Foundation (north-south)	M	SA	R
b. Primary Containment Foundation (east-west)	M	SA	R
c. Primary Containment Foundation (vertical)	M	SA	R

136

METEOROLOGICAL MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.3 The meteorological monitoring instrumentation channels shown in Table 3.3.7.3-1 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the required meteorological monitoring instrumentation channels inoperable for more than 7 days, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrumentation to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.3 Each of the above required meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.3-1.

TABLE 3.3.7.3-1

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM INSTRUMENTS OPERABLE
a. Wind Speed	
1. Elev. 33 ft.	1
2. Elev. 150 ft.	1
b. Wind Direction	
1. Elev. 30 ft.	1
2. Elev. 150 ft.	1
c. Air Temperature Difference	
1. Elev. 150-33 ft.	1

TABLE 4.3.7.3-1

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
a. Wind Speed		
1. Elev. 33 ft.	D	SA
2. Elev. 150 ft.	D	SA
b. Wind Direction		
1. Elev. 33 ft.	D	SA
2. Elev. 150 ft.	D	SA
c. Air Temperature Difference		
1. Elev. 150-33 ft.	D	SA

BEST COPY AVAILABLE

REMOTE SHUTDOWN SYSTEM INSTRUMENTATION AND CONTROLS

LIMITING CONDITION FOR OPERATION

3.3.7.4 The remote shutdown system instrumentation and controls shown in Table 3.3.7.4-1 and Table 3.3.7.4-2 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring instrumentation channels less than required by Table 3.3.7.4-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE remote shutdown system controls less than required in Table 3.3.7.4-2, restore the inoperable control(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.4.1 Each of the above required remote shutdown monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.4-1.

4.3.7.4.2 At least one of each of the above remote shutdown control switch(es) and control circuits shall be demonstrated OPERABLE by verifying its capability to perform its intended function(s) at least once per 18 months.

140

TABLE 3.3.7.4-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM INSTRUMENTS OPERABLE*
1. Reactor Vessel Pressure	2
2. Reactor Vessel Water Level	2
3. Safety/Relief Valve Position, (3) valves	1/valve
4. Suppression Chamber Water Level	2
5. Suppression Chamber Water Temperature	2
6. RHR System Flow	1
7. Safety Auxiliaries Cooling System Flow	1
8. Safety Auxiliaries Cooling System Temperature	1
9. RCIC System Flow	1
10. RCIC Turbine Speed	1
11. RCIC Turbine Bearing Oil Pressure Low Indication	1
12. RCIC High Pressure/Low Pressure Turbine Bearing Temperature High Indication	1
13. Condensate Storage Tank Level Low-Low Indication	1
14. Standby Diesel Generator 1AG400 Breaker Indication	1

---

*Either primary location (Remote Shutdown Panel, 10C399) or alternate location.

141

TABLE 3.3.7.4-1 (Continued)

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT (Continued)	MINIMUM INSTRUMENTS OPERABLE*
15. Standby Diesel Generator 1BG400 Breaker Indication	1
16. Standby Diesel Generator 1CG400 Breaker Indication	1
17. Standby Diesel Generator 1DG400 Breaker Indication	1
18. Switchgear Room Cooler 1AVH401 Status Indication	1
19. Switchgear Room Cooler 1BVH401 Status Indication	1
20. Switchgear Room Cooler 1CVH401 Status Indication	1
21. Switchgear Room Cooler 1DVH401 Status Indication	1

---

*Either primary location (Remote Shutdown Panel 10C399) or alternate location.

142

TABLE 3.3.7.4-2

REMOTE SHUTDOWN SYSTEMS CONTROLS

CHANNEL TRANSFER SWITCHES - REMOTE SHUTDOWN PANEL (RSP)(1)

1SV-HSS-4410A	Control -	Class 1E Channel A Transfer Switch
1SV-HSS-4410B	Control -	Class 1E Channel B Transfer Switch
1SV-HSS-4410C	Control -	Class 1E Channel C Transfer Switch
1SV-HSS-4410D	Control -	Class 1E Channel D Transfer Switch
1SV-HSS-4410N	Control -	Non-Class 1E Transfer Switch

RCIC SYSTEM - RSP

1FC-HV-4282	Control -	RCIC Turbine Trip/Throttle Valve
1FC-HV-F045	Control -	RCIC Turbine Shutoff Valve
1FC-HV-F008	Control -	RCIC Steam Supply Outboard Isolation Valve
1FC-HV-F007	Control -	RCIC Steam Supply Inboard Isolation Valve
1BD-HV-F031	Control -	Suppression Pool to RCIC Pump Suction Valve
1BD-HV-F010	Control -	Condensate Storage Tank to RCIC Pump Suction Valve
1BD-SV-F019	Control -	RCIC Pump Discharge Minimum Flow Valve
1BD-HV-F046	Control -	RCIC Turbine Cooling Water Supply Valve
1BD-HV-F013	Control -	RCIC Pump Discharge to Feedwater Line Isolation Valve
1FC-HV-F076	Control-	RCIC Steam Line Inboard Isolation Valve
1BD-HV-F012(2)	Indication -	RCIC Pump Discharge Valve
1BD-HV-F022(3)	Indication -	Test Return Valve to Condensate Storage Tank
1FC-HV-F059(2)	Indication -	RCIC Turbine Exhaust to Suppression Pool Valve
1FC-HV-F060(2)	Indication -	RCIC Condenser Vacuum Pump Discharge Valve
1FC-HV-F062(2)	Indication -	RCIC Turbine Exhaust Outboard Vacuum Breaker Isolation Valve
1FC-HV-F084(2)	Indication -	RCIC Turbine Exhaust Inboard Vacuum Breaker Isolation Valve
1FC-HV-F025(3)	Indication -	RCIC Condensate Pot Drain to Main Condenser Valve
1FC-HV-F004(3)	Indication -	RCIC Vacuum Tank Condensate Pump Discharge to Clean Rad Waste Valve
1BD-BP228(4)	Indication -	ECCS (RCIC) Jockey Pump BP228
1FC-OP220	Control -	RCIC Vacuum Tank Condensate Pump OP220
1FC-OP219	Control -	RCIC Gland Seal Condenser Vacuum Pump OP219
1FC-FIC-4158	Control -	RCIC System Injection Flow

RHR SYSTEM - RSP

1BC-HV-F006B	Control -	RHR Pump BP202 Suction From Recirc Line Valve
1BC-HV-F004B	Control -	RHR Pump BP202 Suction From Suppression Pool Valve

TABLE 3.3.7.4-2 (Continued)

REMOTE SHUTDOWN SYSTEMS CONTROLS

RHR SYSTEM - RSP (Cont.)

1BC-HV-F007B	Control -	RHR Pump BP202 Minimum Flow Valve to Suppression Pool
1BC-HV-F048B	Control -	RHR Loop B Heat Exchanger Bypass Valve
1BC-HV-F015B	Control -	RHR Loop B Shutdown Cooling Return Valve
1BC-HV-F022	Control -	RHR Reactor Head Spray Inboard Isolation Valve
1BC-HV-F023	Control -	RHR Reactor Head Spray Outboard Isolation Valve
1BC-HV-F009	Control -	RHR Shutdown Cooling Suction From Recirc Line Inboard Isolation Valve
1BC-HV-F008	Control -	RHR Shutdown Cooling Suction From Recirc Line Outboard Isolation Valve
1BC-HV-F122B	Control -	RHR Loop B Shutdown Cooling Injection Check Valve Bypass Valve
1BC-HV-4439	Control -	RHR Discharge to Liquid Radwaste Reactor Building Isolation Valve
1BC-HV-F024B	Control -	RHR Pump BP202 Test Return Valve to Suppression Pool
1BC-HV-F047B	Control -	RHR Loop B Heat Exchanger Shell Side Inlet Valve
1BC-HV-F003B	Control -	RHR Loop B Heat Exchanger Shell Side Outlet Valve
1BC-HV-F049	Control -	RHR Discharge to Liquid Radwaste Inboard Isolation Valve
1BC-HV-F040	Control -	RHR Discharge to Liquid Radwaste Outboard Isolation Valve
1BC-HV-F006A(3)	Indication -	RHR Pump AP202 Suction From Recirc Line Valve
1BC-HV-F010B(3)	Indication -	RHR Pump DP202 Test Return Valve to Suppression Pool
1BC-HV-F016B(3)	Indication -	RHR Loop B Containment Spray Outboard Isolation Valve
1BC-HV-F027B(3)	Indication -	RHR Loop B Suppression Pool Spray Line Isolation Valve
1BC-HV-F017B(3)	Indication -	RHR Low Pressure Coolant Injection Loop B Injection Valve
1BC-HV-F004D(2)	Indication -	RHR Pump DP202 Suction From Suppression Pool Valve
1BC-HV-F021A(3)	Indication -	RHR Loop A Containment Spray Inboard Isolation Valve
1BC-HV-F021B(3)	Indication -	RHR Loop B Containment Spray Inboard Isolation Valve
1BC-BP202	Control -	RHR Pump BP202
1BC-HSS-4416B	Control -	Transfer Switch For RHR Pump BP202
1BC-DP228(4)	Indication -	ECCS (RHR B) Jockey Pump DP228

TABLE 3.3.7.4-2 (Continued)

REMOTE SHUTDOWN SYSTEMS CONTROLS

RHR SYSTEM - REDUNDANT CONTROLS

1BC-HV-F006A	Local Control -	RHR Pump AP202 Suction From Recirc Line Valve
1BC-HV-F004A	Local Control -	RHR Pump AP202 Suction From Suppression Pool Valve
1BC-HV-F048A	Local Control -	RHR Loop A Heat Exchanger Bypass Valve
1BC-HV-F015A	Local Control -	RHR Loop A Shutdown Cooling Return Valve
1BC-HV-F024A	Local Control -	RHR Pump AP202 Test Return Valve to Suppression Pool
1BC-HV-F047A	Local Control -	RHR Loop A Heat Exchanger Shell Side Inlet Valve
1BC-HV-F003A	Local Control -	RHR Loop A Heat Exchanger Shell Side Outlet Valve
1BC-AP202	Local Control -	RHR Pump AP202

SACS - RSP

1EG-HV-2522B/ 1EG-HV-2496B(5)	Control -	SACS Loop B to Turbine Auxiliaries Cooling System (TACS) Inboard Supply and Return Valves
1EG-HV-2522D/ 1EG-HV-2496D(6)	Control -	SACS Loop B to TACS Outboard Supply and Return Valves
1EG-HV-2512B	Control -	RHR Loop B Heat Exchanger Tube Side Outlet Valve
1EG-HV-2491B	Control -	SACS Loop B Heat Exchanger B1E201, Inlet Valve
1EG-HV-2494B	Control -	SACS Loop B Heat Exchanger B2E201, Inlet Valve
1EG-HV-2520B(7)(2)	Indication -	RHR Pump BP202 Seal and Motor Bearing Coolers Cooling Water Supply Valve
1EG-BP210	Control -	SACS Loop B Pump BP210
1EG-HSS-2485B	Control -	Transfer Switch For SACS Loop B Pump BP210
1EG-DP210	Control -	SACS Loop B Pump DP210
1EG-HSS-2485D	Control -	Transfer Switch For SACS Loop B Pump DP210

SACS - REDUNDANT CONTROLS

1EG-HV-2496A	Local Control -	SACS Loop A Return From TACS Inboard Valve
1EG-HV-2496C	Local Control -	SACS Loop A Return From TACS Outboard Valve
1EG-HV-2512A	Local Control -	RHR Loop A Heat Exchanger Tube Side Outlet Valve

TABLE 3.3.7.4-2 (Continued)

REMOTE SHUTDOWN SYSTEMS CONTROLS

SACS - REDUNDANT CONTROLS (Cont.)

1EG-AP210	Local Control -	SACS Loop A Pump AP210
1EG-CP210	Local Control -	SACS Loop A Pump CP210

STATION SERVICE WATER SYSTEM (SSWS) - RSP

1EA-HV-2204	Control -	Reactor Auxiliaries Cooling System (RACS) Heat Exchanger Supply Valve (From SACS Loop B)
1EA-HV-2355B	Control -	SACS Loop B Heat Exchanger B2E201 Outlet Valve
1EA-HV-2371B	Control -	SACS Loop B Heat Exchanger B1E201 Outlet Valve
1EA-HV-2357B	Control -	SACS Loop B to Cooling Tower Valve
1EA-HV-2198B	Control -	SSWS Pump BP502 Discharge Valve
1EA-HV-2198D	Control -	SSWS Pump DP502 Discharge Valve
1EA-HV-2197B	Control -	SSWS Strainer BF509 Main Backwash Valve
1EA-HV-2197D	Control -	SSWS Strainer DF509 Main Backwash Valve
1EA-BP502	Control -	SSWS Pump BP502
1EA-HSS-2219B	Control -	Transfer Switch For SSWS Pump BP502
1EA-DP502	Control -	SSWS Pump DP502
1EA-HSS-2219D	Control -	Transfer Switch For SSWS Pump DP502

SSWS - REDUNDANT CONTROLS

1EA-HV-2203	Local Control -	RACS Heat Exchanger Supply Valve (From SACS Loop A)
1EA-AP502	Local Control -	SSWS Pump AP502
1EA-CP502	Local Control -	SSWS Pump CP502

CONTROL AREA CHILLED WATER SYSTEM (CACWS) - RSP

1GJ-BK400	Control -	Control Area Chiller BK400
1GJ-HSS-9652B	Control -	Transfer Switch For Control Area Chiller BK400
1GJ-BK403	Control -	Safety-Related Panel Room Chiller BK403
1GJ-HSS-9666B4	Control -	Transfer Switch For Safety-Related Panel Room Chiller BK403
1GJ-BP400	Control -	Control Area Chilled Water Circulating Pump BP400
1GJ-BP414	Control -	Safety-Related Panel Room Chilled Water Circulating Pump BP414

TABLE 3.3.7.4-2 (Continued)

REMOTE SHUTDOWN SYSTEMS CONTROLS

CACWS - REDUNDANT CONTROLS

1GJ-AK400	Local Control -	Control Area Chiller AK400
1GJ-AK403	Local Control -	Safety-Related Panel Room Chiller AK403
1GJ-AP400	Local Control -	Control Area Chilled Water Circulating Pump AP400
1GJ-AP414	Local Control -	Safety-Related Panel Room Chilled Water Circulating Pump AP414

REACTOR RECIRCULATION SYSTEM -RSP

1BB-HV-F031B(3)	Indication -	Reactor Recirculation Pump BP201 Discharge Valve

SAFETY/RELIEF VALVES - RSP

1AB-PSV-F013F	Control -	Main Steam Line B Safety/Relief Valve
1AB-PSV-F013H	Control -	Main Steam Line D Safety/Relief Valve
1AB-PSV-F013M	Control -	Main Steam Line D Safety/Relief Valve

SAFETY/RELIEF VALVES - REDUNDANT CONTROLS

1AB-PSV-F013A	Local Control -	Main Steam Line A Safety/Relief Valve
1AB-PSV-F013E	Local Control -	Main Steam Line C Safety/Relief Valve

---

(1) The Remote Shutdown Panel (RSP) is Panel 10C399.
(2) Valve is signalled to open on RSP Takeover.
(3) Valve is signalled to close on RSP Takeover.
(4) Pump is signalled to run on RSP Takeover.
(5) Operation of valve 1EG-HV-2496B is ganged to operation of valve 1EG-HV-2522B.
(6) Operation of valve 1EG-HV-2496D is ganged to operation of valve 1EG-HV-2522D.
(7) Operation of valve 1EG-HV-2520B is ganged to operation of RHR Pump BP202.

TABLE 4.3.7.4-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Vessel Pressure	M	R
2. Reactor Vessel Water Level	M	R
3. Safety/Relief Valve Position (Energization)	M	NA
4. Suppression Chamber Water Level	M	R
5. Suppression Chamber Water Temperature	M	R
6. RHR System Flow	M	R
7. Safety Auxiliaries Cooling System Flow	M	R
8. Safety Auxiliaries Cooling System Temperature	M	R
9. RCIC System Flow	M	R
10. RCIC Turbine Speed	M	R
11. RCIC Turbine Bearing Oil Pressure Low Indication	M	R
12. RCIC High Pressure/Low Pressure Turbine Bearing Temperature High Indication	M	R

TABLE 4.3.7.4-1 (Continued)

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
13. Condensate Storage Tank Level Low-Low Indication	M	R
14. Standby Diesel Generator 1AG400 Breaker Indication	M	NA
15. Standby Diesel Generator 1BG400 Breaker Indication	M	NA
16. Standby Diesel Generator 1CG400 Breaker Indication	M	NA
17. Standby Diesel Generator 1DG400 Breaker Indication	M	NA
18. Switchgear Room Cooler 1AVH401 Status Indication	M	NA
19 Switchgear Room Cooler 1BVH401 Status Indication	M	NA
20. Switchgear Room Cooler 1CVH401 Status Indication	M	NA
21. Switchgear Room Cooler 1DVH401 Status Indication	M	NA

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.5 The accident monitoring instrumentation channels shown in Table 3.3.7.5-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3.7.5-1.

ACTION:

With one or more accident monitoring instrumentation channels inoperable, take the ACTION required by Table 3.3.7.5-1.

SURVEILLANCE REQUIREMENTS

4.3.7.5 Each of the above required accident monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.5-1.

TABLE 3.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. Reactor Vessel Pressure	2	1	1,2,3	80
2. Reactor Vessel Water Level	2	1	1,2,3	80
3. Suppression Chamber Water Level	2	1	1,2,3	80
4. Suppression Chamber Water Temperature*	2	2	1,2,3	80(a)
5. Suppression Chamber Pressure	2	1	1,2,3	80
6. Drywell Pressure	2	1	1,2,3	80
7. Drywell Air Temperature	2	1	1,2,3	80
8. Primary Containment Hydrogen/Oxygen Concentration Analyzer and Monitor	2	1	1,2,3	80
9. Safety/Relief Valve Position Indicators	2/valve**	1/valve**	1,2,3	80
10. Drywell Atmosphere Post-Accident Radiation Monitor	2	1	1,2,3	81
11. North Plant Vent Radiation Monitor#	1	1	1,2,3	81
12. South Plant Vent Radiation Monitor#	1	1	1,2,3	81
13. FRVS Vent Radiation Monitor#	1	1	1,2,3	81
14. Primary Containment Isolation Valve Position Indication(b)	2/valve	1/valve	1,2,3	82

#High range noble gas monitors.

*Average bulk pool temperature.

**Acoustic monitoring and tail pipe temperature.

(a)Suppression chamber water temperature instrumentation must satisfy the availability requirements specified in Specification 3.6.2.1.

(b)One channel consists of the open limit switch, and the other channel consists of the closed limit switch.

151

Table 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION
ACTION STATEMENTS

ACTION 80 -

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels shown in Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 81 - With the number of OPERABLE accident monitoring instrumentation channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

a. Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

b. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

The provisions of Specification 3.0.4 are not applicable.

ACTION 82 -

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels shown in Table 3.3.7.5-1, verify the valve(s) position by use of alternate indication methods; restore the inoperable channel(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3.7.5-1, verify the valve(s) position by use of alternate indication methods; restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

TABLE 4.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	APPLICABLE OPERATIONAL CONDITIONS
1. Reactor Vessel Pressure	M	R	1,2,3
2. Reactor Vessel Water Level	M	R	1,2,3
3. Suppression Chamber Water Level	M	R	1,2,3
4. Suppression Chamber Water Temperature	M	R	1,2,3
5. Suppression Chamber Pressure	M	R	1,2,3
6. Drywell Pressure	M	R	1,2,3
7. Drywell Air Temperature	M	R	1,2,3
8. Primary Containment Hydrogen/Oxygen Concentration Analyzer and Monitor	M	Q*	1,2,3
9. Safety/Relief Valve Position Indicators	M	R	1,2,3
10. Drywell Atmosphere Post-Accident Radiation Monitor	M	R**	1,2,3
11. North Plant Vent Radiation Monitor#	M	R	1,2,3
12. South Plant Vent Radiation Monitor#	M	R	1,2,3
13. FRVS Vent Radiation Monitor#	M	R	1,2,3
14. Primary Containment Isolation Valve Position Indication	M	R	1,2,3

---

*Using sample gas containing:
a. Five volume percent oxygen balance nitrogen (oxygen analyzer channel).
b. Five volume percent hydrogen, balance nitrogen (hydrogen analyzer channel).

**CHANNEL CALIBRATION shall consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr and a one point calibration check of the detector below 10 R/hr with an installed or portable gamma source.

#High range noble gas monitors.

SOURCE RANGE MONITORS

LIMITING CONDITION FOR OPERATION

3.3.7.6 At least the following source range monitor channels shall be OPERABLE:

a. In OPERATIONAL CONDITION 2*, three.

b. In OPERATIONAL CONDITION 3 and 4, two.

APPLICABILITY: OPERATIONAL CONDITIONS 2*, 3 and 4.

ACTION:

a. In OPERATIONAL CONDITION 2* with one of the above required source range monitor channels inoperable, restore at least 3 source range monitor channels to OPERABLE status within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4 with one or more of the above required source range monitor channels inoperable, verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.

SURVEILLANCE REQUIREMENTS

4.3.7.6 Each of the above required source range monitor channels shall be demonstrated OPERABLE by:

a. Performance of a:

1. CHANNEL CHECK at least once per:

   a) 12 hours in CONDITION 2*, and

   b) 24 hours in CONDITION 3 or 4.

2. CHANNEL CALIBRATION** at least once per 18 months.

b. Performance of a CHANNEL FUNCTIONAL TEST:

1. Within 24 hours prior to moving the reactor mode switch from the Shutdown position, if not performed within the previous 7 days, and

2. At least once per 31 days.

c. Verifying, prior to withdrawal of control rods, that the SRM count rate is at least 3 cps*** with the detector fully inserted.

---

*With IRM's on range 2 or below.

**Neutron detectors may be excluded from CHANNEL CALIBRATION.

***May be reduced to 0.7 cps provided the signal-to-noise ratio is ≥ 2.

TRAVERSING IN-CORE PROBE SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.7. The traversing in-core probe system shall be OPERABLE with:

a. Five movable detectors, drives and readout equipment to map the core, and

b. Indexing equipment to allow all five detectors to be calibrated in a common location.

APPLICABILITY: When the traversing in-core probe is used for:

a. Recalibration of the LPRM detectors, and

b.* Monitoring the APLHGR, LHGR, MCPR, or MFLPD.

ACTION:

With the traversing in-core probe system inoperable, suspend use of the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.7 The traversing in-core probe system shall be demonstrated OPERABLE by normalizing each of the above required detector outputs within 72 hours prior to use for the LPRM calibration function.

---

*Only the detector(s) in the required measurement location(s) are required to be OPERABLE.

155

LOOSE-PART DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.8 The loose-part detection system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one or more loose-part detection system channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.8 Each channel of the loose-part detection system shall be demonstrated OPERABLE by performance of a:

a. CHANNEL CHECK at least once per 24 hours,

b. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

c. CHANNEL CALIBRATION at least once per 18 months.

BEST COPY AVAILABLE

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.9 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3.7.9-1 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The Alarm/Trip Setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3.7.9-1. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION, or explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.7 why this inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.9 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST at the frequencies shown in Table 4.3.7.9-1.

157

TABLE 3.3.7.9-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste Discharge Line to the Cooling Tower Blowdown Line	1	110
2. RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Cooling Tower Blowdown Effluent	1	111
3. FLOW RATE MEASUREMENT DEVICES		
a. Liquid Radwaste Discharge Line to Cooling Tower Blowdown Line	1	112
b. Cooling Tower Blowdown Weir	1	112

158

TABLE 3.3.7.9-1 (Continued)

TABLE NOTATION

ACTION 110 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.2, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 111 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided that, at least once per 12 hours, grab samples are collected and analyzed for gross radioactivity at a limit of detection of at least 10^{-7} microcuries/ml. Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 112 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves generated in place may be used to estimate flow.

159

TABLE 4.3.7.9-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Discharge Line to the Cooling Tower Blowdown Line	D	P	R(3)	Q(1)
2. RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE				
a. Cooling Tower Blowdown Effluent	D	M	R(3)	Q(2)
3. FLOW RATE MEASUREMENT DEVICES				
a. Liquid Radwaste Discharge Line to Cooling Tower Blowdown Line	D(4)	N.A.	R	Q
b. Cooling Tower Blowdown Weir	D(4)	N.A.	R	Q

BEST COPY

TABLE 4.3.7.9-1 (Continued)

TABLE NOTATIONS

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exists:

   a. Instrument indicates measured levels above the Alarm/Trip Setpoint, or

   b. Circuit failure, or

   c. Instrument indicates a downscale failure.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

   a. Instrument indicates measured levels above the Alarm Setpoint, or

   b. Circuit failure, or

   c. Instrument indicates a downscale failure.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration or are NBS traceable shall be used.

(4) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

BEST COPY AVAILABLE

161

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.10 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3.7.10-1 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specifications 3.11.2.1 and 3.11.2.6 are not exceeded. The Alarm/Trip Setpoints of these channels meeting Specification 3.11.2.1 shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 3.3.7.10-1.

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3.7.10-1. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION, or explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.7 why this inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.10 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies shown in Table 4.3.7.10-1.

162

TABLE 3.3.7.10-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
a. Hydrogen Monitor	1	**	124
2. FILTRATION, RECIRCULATION AND VENTILATION MONITORING SYSTEM			
a. Noble Gas Activity Monitor	1	*	123
b. Iodine Sampler	1	*	125
c. Particulate Sampler	1	*	125
d. Flow Rate Monitor	1	*	122
e. Sampler Flow Rate Monitor	1	*	122
3. SOUTH PLANT VENT MONITORING SYSTEM			
a. Noble Gas Activity Monitor	1	*	123
b. Iodine Sampler	1	*	125
c. Particulate Sampler	1	*	125
d. Flow Rate Monitor	1	*	122
e. Sampler Flow Rate Monitor	1	*	122

TABLE 3.3.7.10-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
4.	NORTH PLANT VENT MONITORING SYSTEM			
	a. Noble Gas Activity Monitor	1	*	123
	b. Iodine Sampler	1	*	125
	c. Particulate Sampler	1	*	125
	d. Flow Rate Monitor	1	*	122
	e. Sampler Flow Rate Monitor	1	*	122

TABLE 3.3.7.10-1 (Continued)

TABLE NOTATION

* At all times.

** During operation of the main condenser air ejector.

ACTION 122 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours. Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 123 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 12 hours and these samples are analyzed for gross activity within 24 hours. Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 124 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of main condenser offgas treatment system may continue for up to 30 days provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours. Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 125 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that within 8 hours samples are continuously collected with auxiliary sampling equipment as required in Table 4.11.2.1.2-1.

165

TABLE 4.3.7.10-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
a. Hydrogen Monitor	D	N.A.	Q(3)	M	**
2. FILTRATION, RECIRCULATION AND VENTILATION MONITORING SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(2)	Q(1)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
3. SOUTH PLANT VENT MONITORING SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(2)	Q(1)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

TABLE 4.3.7.10-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. NORTH PLANT VENT MONITORING SYSTEM					
a. Noble Gas Actvity Monitor	D	M	R(2)	Q(1)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

167

TABLE 4.3.7.10-1 (Continued)

TABLE NOTATION

* At all times.

** During operation of the main condenser air ejector.

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.

(2) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration or are NBS traceable shall be used.

(3) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. Zero volume percent hydrogen, balance nitrogen, and
2. 1.5 volume percent hydrogen, balance nitrogen.

168

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.8 At least one turbine overspeed protection system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one turbine control valve, or one main stop valve per high pressure turbine steam lead inoperable and/or with one combined intermediate valve per low pressure turbine steam lead inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours or close at least one valve in the affected steam lead(s) or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required turbine overspeed protection system otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.8.1 The provisions of Specification 4.0.4 are not applicable.

4.3.8.2 The above required turbine overspeed protection system shall be demonstrated OPERABLE:

a. At least once per 7 days by:

   1. Cycling each of the following valves through at least one complete cycle from the running position:

      a) For the overspeed protection control system;

         1) Six low pressure combined intermediate valves

      b) For the electrical overspeed trip system and the mechanical overspeed trip system;

         1) Four high pressure main stop valves, and

         2) Six low pressure combined intermediate valves.

169

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by:

  1. Cycling each of the following valves through at least one complete cycle from the running position:

    a) For the overspeed protection control system;

      1) Four high pressure turbine control valves

    b) For the electrical overspeed trip system and the mechanical overspeed trip system;

      1) Four high pressure turbine control valves.

c. At least once per 18 months by performance of a CHANNEL CALIBRATION of the turbine overspeed protection instrumentation.

d. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of all valve seats, disks and stems and verifying no unacceptable flaws or excessive corrosion. If unacceptable flaws or excessive corrosion are found, all other valves of that type shall be inspected.

170

3/4.3.9 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.9 The feedwater/main turbine trip system actuation instrumentation channels shown in Table 3.3.9-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.9-2.

APPLICABILITY: As shown in Table 3.3.9-1.

ACTION:

a. With a feedwater/main turbine trip system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.9-2, declare the channel inoperable and either place the inoperable channel in the tripped condition until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value, or declare the associated system inoperable.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 7 days or be in at least STARTUP within the next 6 hours.

c. With the number of OPERABLE channels two less than required by the Minimum OPERABLE Channels requirement, restore at least one of the inoperable channels to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.9.1 Each feedwater/main turbine trip system actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.9.1-1.

4.3.9.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

TABLE 3.3.9-1

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNIT	MININMUM OPERABLE CHANNELS	APPLICABLE OPERATIONAL CONDITIONS
1. Reactor Vessel Water Level-High, Level 8	3	1

TABLE 3.3.9-2

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel Water Level-High, Level 8	≤ 54.0 inches*	≤ 55.5 inches

---

*See Bases Figure B 3/4 3-1.

TABLE 4.3.9.1-1

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTIONAL CHECK	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Reactor Vessel Water Level-High, Level 8	S	M	R	1

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation with:

a. Total core flow greater than or equal to 45% of rated core flow, or

b. THERMAL POWER less than or equal to the limit specified in Figure 3.4.1.1-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation, immediately initiate action to reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours and initiate measures to place the unit in at least HOT SHUTDOWN within 12 hours.

b. With no reactor coolant system recirculation loops in operation, immediately initiate action to reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours and initiate measures to place the unit in at least STARTUP within 6 hours and in HOT SHUTDOWN within the next 6 hours.

c. With two reactor coolant system recirculation loops in operation and total core flow less than 45% of rated core flow and THERMAL POWER greater than the limit specified in Figure 3.4.1.1-1:

    1. Determine the APRM and LPRM** noise levels (Surveillance 4.4.1.1.3):

        a) At least once per 8 hours, and

        b) Within 30 minutes after the completion of a THERMAL POWER increase of at least 5% of RATED THERMAL POWER.

    2. With the APRM or LPRM** neutron flux noise levels greater than three times their established baseline noise levels, immediately initiate corrective action to restore the noise levels to within the required limits within 2 hours by increasing core flow to greater than 45% of rated core flow or by reducing THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1.

---

*See Special Test Exception 3.10.4.

**Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.

175

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 At least once per 12 hours verify that total core flow is greater than or equal to 45% of rated core flow and/or that THERMAL POWER is less than the limit specified in Figure 3.4.1.1-1.

4.4.1.1.2 Each pump MG set scoop tube mechanical and electrical stop shall be demonstrated OPERABLE with overspeed setpoints less than or equal to 105% and 102.5%, respectively, of rated core flow, at least once per 18 months.

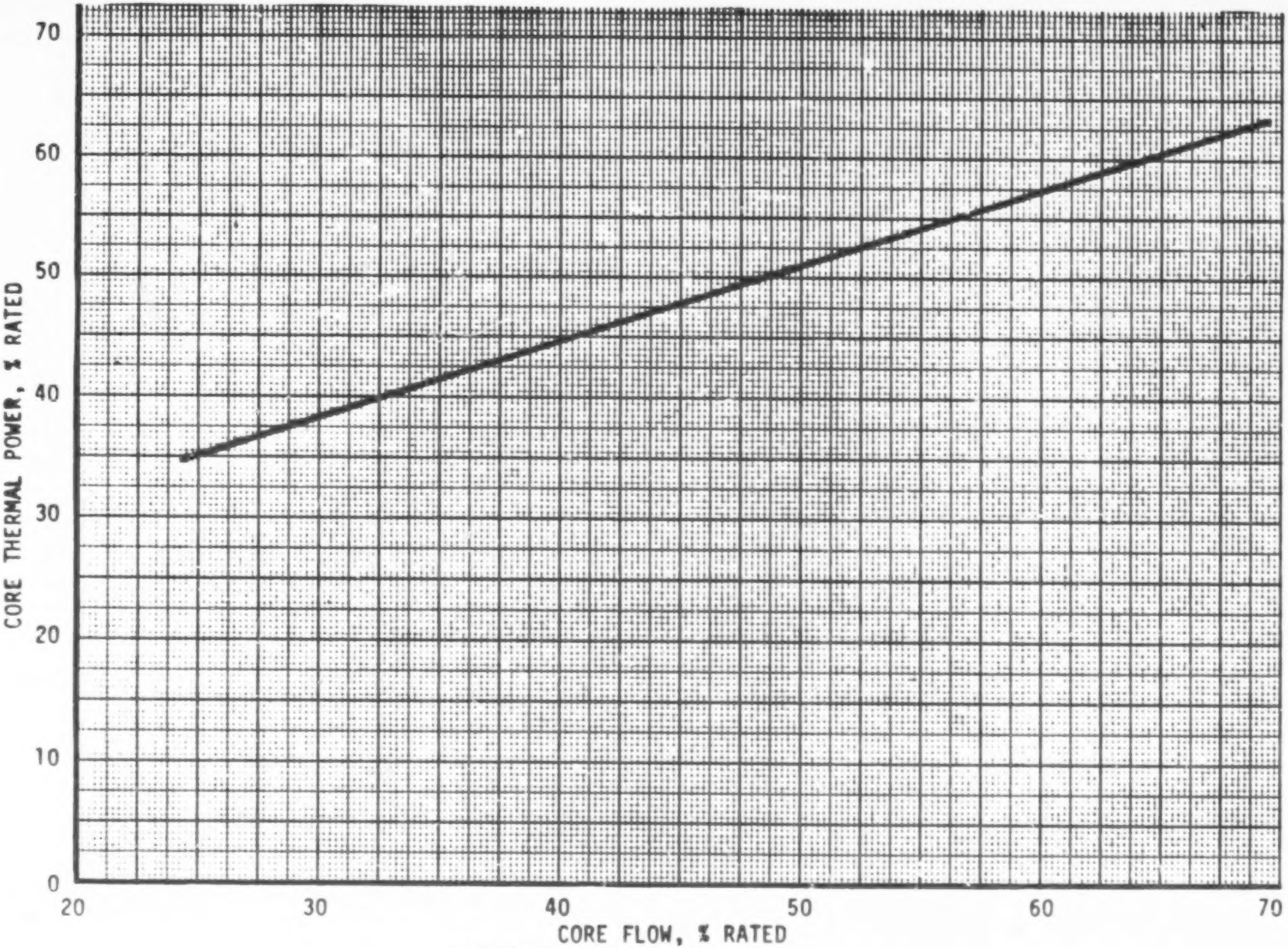
4.4.1.1.3 Establish a baseline APRM and LPRM** neutron flux noise value within the regions for which monitoring is required (Specification 3.4.1.1, ACTION c) within 2 hours of entering the region for which monitoring is required unless baselining has previously been performed in the region since the last refueling outage.

---

*If not performed within the previous 31 days.

**Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.

176



THERMAL POWER VERSUS CORE FLOW
FIGURE 3.4.1.1-1

JET PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one or more jet pumps inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS*

4.4.1.2 Each of the above required jet pumps shall be demonstrated OPERABLE prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER and at least once per 24 hours by determining recirculation loop flow, total core flow and diffuser-to-lower plenum differential pressure for each jet pump and verifying that no two of the following conditions occur when the recirculation pumps are operating in accordance with Specification 3.4.1.3.

a. The indicated recirculation loop flow differs by more than 10% from the established pump speed-loop flow characteristics.

b. The indicated total core flow differs by more than 10% from the established total core flow value derived from recirculation loop flow measurements.

c. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from the established patterns by more than 10%.

---

*During the startup test program, data shall be recorded for the parameters listed to provide a basis for establishing the specified relationships. Comparisons of the actual data in accordance with the criteria listed shall commence upon conclusion of the startup test program.

178

RECIRCULATION PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation pump speed shall be maintained within:

a. 5% of each other with core flow greater than or equal to 70% of rated core flow.

b. 10% of each other with core flow less than 70% of rated core flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

With the recirculation pump speeds different by more than the specified limits, either:

a. Restore the recirculation pump speeds to within the specified limit within 2 hours, or

b. Declare the recirculation loop of the pump with the slower speed not in operation and take the ACTION required by Specification 3.4.1.1.

SURVEILLANCE REQUIREMENTS

4.4.1.3 Recirculation pump speed shall be verified to be within the limits at least once per 24 hours

---

*See Special Test Exception 3.10.4.

179

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.4 An idle recirculation loop shall not be started unless the temperature differential between the reactor pressure vessel steam space coolant and the bottom head drain line coolant is less than or equal to 145°F and:

a. When both loops have been idle, unless the temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is less than or equal to 50°F, or

b. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle and operating recirculation loops is less than or equal to 50°F and the operating loop flow rate is less than or equal to 50% of rated loop flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

With temperature differences and/or flow rates exceeding the above limits, suspend startup of any idle recirculation loop.

SURVEILLANCE REQUIREMENTS

4.4.1.4 The temperature differentials and flow rate shall be determined to be within the limits within 15 minutes prior to startup of an idle recirculation loop.

ST COPY [illegible]

3/4.4.2 SAFETY/RELIEF VALVES

SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2.1 The safety valve function of at least 13 of the following reactor coolant system safety/relief valves shall be OPERABLE*# with the specified code safety valve function lift settings:**

4 safety-relief valves @ 1108 psig ±1%
5 safety-relief valves @ 1120 psig ±1%
5 safety-relief valves @ 1130 psig ±1%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the safety valve function of two or more of the above listed fourteen safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 110°F, close the stuck open safety relief valve(s); if unable to close the stuck open valve(s) within 2 minutes or if suppression pool average water temperature is 110°F or greater, place the reactor mode switch in the Shutdown position.

c. With one or more of the above required safety/relief valve acoustic monitors inoperable, restore the inoperable monitors to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

---

*SRVs which perform as ADS function must also satisfy the OPERABILITY requirements of Specification 3.5.1, ECCS-Operating.

**The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

#SRVs which perform a low-low set function must also satisfy the OPERABILITY requirements of Specification 3.2.2, Safety/Relief Valves Low-Low Set Function.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.2.1 The acoustic monitor for each safety/relief valve shall be demonstrated OPERABLE with the setpoint verified to be $\leq$ 30% of full open noise level** by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and a

b. CHANNEL CALIBRATION at least once per 18 months*.

4.4.2.2 At least 1/2 of the safety relief valves shall be removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations at least once per 18 months, and they shall be rotated such that all 14 safety relief valves are removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations at least once per 40 months.

---

*The provisions of Specification 4.0.4 are not applicable provided the Surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

**Initial setting shall be in accordance with the manufacturer's recommendations. Adjustment to the valve full open noise level shall be accomplished during the startup test program.

REACTOR COOLANT SYSTEM

SAFETY/RELIEF VALVES LOW-LOW SET FUNCTION

LIMITING CONDITION FOR OPERATION

3.4.2.2 The relief valve function and the low-low set function of the following reactor coolant system safety/relief valves shall be OPERABLE with the following settings:

Valve No.	Low-Low Set Function Setpoint* (psig) ±2%	
	Open	Close
F013H	1017	905
F013P	1047	935

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the relief valve function and/or the low-low set function of one of the above required reactor coolant system safety/relief valves inoperable, restore the inoperable relief valve function and low-low set function to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the relief valve function and/or the low-low set function of both of the above required reactor coolant system safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2.1 The relief valve function and the low-low set function pressure actuation instrumentation shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days.

b. CHANNEL CALIBRATION, LOGIC SYSTEM FUNCTIONAL TEST and simulated automatic operation of the entire system (excluding actual valve actuation) at least once per 18 months.

---

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.3.1 The following reactor coolant system leakage detection systems shall be OPERABLE:

a. The drywell atmosphere gaseous radioactivity monitoring system,*
b. The drywell floor and equipment drain sump monitoring system,
c. The drywell air cooler condensate flow rate monitoring system,
d. The drywell pressure monitoring system, and
e. The drywell temperature monitoring system.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With only four of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required drywell atmosphere gaseous radioactivity monitoring system, the drywell pressure monitoring system, the drywell temperature monitoring system and/or the drywell air cooler condensate flow rate monitoring system is inoperable; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

a. Drywell atmosphere gaseous radioactivity monitoring system-performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

b. The drywell pressure shall be monitored at least once per 12 hours and the drywell temperature shall be monitored at least once per 24 hours.

c. Drywell floor and equipment drain sump monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION TEST at least once per 18 months.

d. Drywell air coolers condensate flow rate monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

---

* Not required to be OPERABLE prior to 150 days after initial fuel load.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm IDENTIFIED LEAKAGE averaged over any 24-hour period.

d. 0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1, at rated pressure.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one other closed manual or deactivated automatic or check* valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one or more of the high/low pressure interface valve leakage pressure monitors shown in Table 3.4.3.2-2 inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm setpoint at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

---

*Which have been verified not to exceed the allowable leakage limit at the last refueling outage or the after last time the valve was disturbed, whichever is more recent.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the drywell atmospheric gaseous radioactivity at least once per 12 hours (not a means of quantifying leakage),

b. Monitoring the drywell floor and equipment drain sump flow rate at least once per 12 hours, and

c. Monitoring the drywell air coolers condensate flow rate at least once per 12 hours, and

d. Monitoring the drywell pressure at least once per 12 hours (not a means of quantifying leakage), and

e. Monitoring the reactor vessel head flange leak detection system at least once per 24 hours (not a means of quantifying leakage), and

f. Monitoring the drywell temperature at least once per 24 hours (not a means of quantifying leakage).

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 and verifying the leakage of each valve to be within the specified limit:

a. At least once per 18 months, and

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

4.4.3.2.3 The high/low pressure interface valve leakage pressure monitors shall be demonstrated OPERABLE with alarm setpoints per Table 3.4.3.2-2 by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

b. CHANNEL CALIBRATION at least once per 18 months.

TABLE 3.4.3.2-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

1ST ISOLATION VALVE(S) NUMBERS(S)	2ND ISOLATION VALVE(S) NUMBER(S)	PRESSURE INDICATION	SERVICE
BE-V006 BE-V071	BE-V007	1-BE-PISH-N654A	'A' Core Spray/ HPCI Injection
BE-V002 BE-V072	BE-V003	1-BE-PISH-N654B	'B' Core Spray Injection
BC-V114 BC-V119	BC-V113	1-BC-PISH-N653A	'A' LPCI Injection
BC-V017 BC-V120	BC-V016	1-BC-PISH-N653B	'B' LPCI Injection
BC-V102 BC-V121	BC-V101	1-BC-PISH-N653C	'C' LPCI Injection
BC-V005 BC-V122	BC-V004	1-BC-PISH-N653D	'D' LPCI Injection
BC-V021	BC-V020	1-BC-PISH-N653B	Head Spray
BC-V111 BC-V117	BC-V110	1-BC-PISH-N653A	'A' Shutdown Cooling Return to 'A' Recirc Loop
BC-V014 BC-V118	BC-V013	1-BC-PISH-N653B	'B' Shutdown Cooling Return to 'B' Recirc Loop
BC-V071	BC-V164	1-BC-PISH-N657	Shutdown Cooling Supply From 'B' Recirc Loop

TABLE 3.4.3.2-2

REACTOR COOLANT SYSTEM INTERFACE VALVES

LEAKAGE PRESSURE MONITORS

SERVICE	INSTRUMENT	ALARM SETPOINT (psig)	ALARM ALLOWABLE VALUE (psig)
Core Spray	1-BE-PISH-N654A	475	≤500
Core Spray	1-BE-PISH-N654B	475	≤500
LPCI/RHR	1-BC-PISH-N653A	380	≤410
LPCI/RHR	1-BC-PISH-N653B	380	≤410
LPCI/RHR	1-BC-PISH-N653C	380	≤410
LPCI/RHR	1-BC-PISH-N653D	380	≤410
RHR	1-BC-PISH-N657	130	≤155

3/4.4.4 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.4 The chemistry of the reactor coolant system shall be maintained within the limits specified in Table 3.4.4-1.

APPLICABILITY: At all times.

ACTION:

a. In OPERATIONAL CONDITION 1:

   1. With the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.4.4-1 for less than 72 hours during one continuous time interval and, for conductivity and chloride concentration, for less than 336 hours per year, but with the conductivity less than 10 μmho/cm at 25°C and with the chloride concentration less than 0.5 ppm, this need not be reported to the Commission and the provisions of Specification 3.0.4 are not applicable.

   2. With the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.4.4-1 for more than 72 hours during one continuous time interval or with the conductivity and chloride concentration exceeding the limit specified in Table 3.4.4-1 for more than 336 hours per year, be in at least STARTUP within the next 6 hours.

   3. With the conductivity exceeding 10 μmho/cm at 25°C or chloride concentration exceeding 0.5 ppm, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. In OPERATIONAL CONDITION 2 and 3 with the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.4.4-1 for more than 48 hours during one continuous time interval, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. At all other times:

   1. With the:

      a) Conductivity or pH exceeding the limit specified in Table 3.4.4-1, restore the conductivity and pH to within the limit within 72 hours, or

      b) Chloride concentration exceeding the limit specified in Table 3.4.4-1, restore the chloride concentration to within the limit within 24 hours, or

      perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system. Determine that the structural integrity of the reactor coolant system remains acceptable for continued operation prior to proceeding to OPERATIONAL CONDITION 3.

   2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4 The reactor coolant shall be determined to be within the specified chemistry limit by:

a. Measurement prior to pressurizing the reactor during each startup, if not performed within the previous 72 hours.

b. Analyzing a sample of the reactor coolant for:

   1. Chlorides at least once per:

      a) 72 hours, and

      b) 8 hours whenever conductivity is greater than the limit in Table 3.4.4-1.

   2. Conductivity at least once per 72 hours.

   3. pH at least once per:

      a) 72 hours, and

      b) 8 hours whenever conductivity is greater than the limit in Table 3.4.4-1.

c. Continuously recording the conductivity of the reactor coolant, or, when the continuous recording conductivity monitor is inoperable, obtaining an in-line conductivity measurement at least once per:

   1. 4 hours in OPERATIONAL CONDITIONS 1, 2 and 3, and

   2. 24 hours at all other times.

d. Performance of a CHANNEL CHECK of the continuous conductivity monitor with an in-line flow cell at least once per:

   1. 7 days, and

   2. 24 hours whenever conductivity is greater than the limit in Table 3.4.4-1.

190

TABLE 3.4.4-1

REACTOR COOLANT SYSTEM CHEMISTRY LIMITS

OPERATIONAL CONDITION	CHLORIDES	CONDUCTIVITY (µmhos/cm @25°C)	PH
1	$\leq$ 0.2 ppm	$\leq$ 1.0	$5.6 \leq \text{pH} \leq 8.6$
2 and 3	$\leq$ 0.1 ppm	$\leq$ 2.0	$5.6 \leq \text{pH} \leq 8.6$
At all other times	$\leq$ 0.5 ppm	$\leq$ 10.0	$5.3 \leq \text{pH} \leq 8.6$

3/4.4.5 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.5 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 0.2 microcuries per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 100/Ē microcuries per gram.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

a. In OPERATIONAL CONDITIONS 1, 2 or 3 with the specific activity of the primary coolant;

  1. Greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or greater than 4.0 microcuries per gram DOSE EQUIVALENT I-131, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.

  2. Greater than 100/Ē microcuries per gram, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.

b. In OPERATIONAL CONDITIONS 1, 2, 3 or 4, with the specific activity of the primary coolant greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 or greater than 100/Ē microcuries per gram, perform the sampling and analysis requirements of Item 4a of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit.

c. In OPERATIONAL CONDITION 1 or 2, with:

  1. THERMAL POWER changed by more than 15% of RATED THERMAL POWER in one hour*, or

  2. The off-gas level, at the SJAE, increased by more than 10,000 microcuries per second in one hour during steady state operation at release rates less than 75,000 microcuries per second, or

  3. The off-gas level, at the SJAE, increased by more than 15% in one hour during steady state operation at release rates greater than 75,000 microcuries per second,

  perform the sampling and analysis requirements of Item 4b of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit.

---

*Not applicable during the startup test program.

192

SURVEILLANCE REQUIREMENTS

4.4.5 The specific activity of the reactor coolant shall be demonstrated to be within the limits by performance of the sampling and analysis program of Table 4.4.5-1.

193

TABLE 4.4.5-1

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	OPERATIONAL CONDITIONS IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Beta and Gamma Activity Determination	At least once per 72 hours	1, 2, 3
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	At least once per 31 days	1
3. Radiochemical for $\bar{E}$ Determination	At least once per 6 months*	1
4. Isotopic Analysis for Iodine	a) At least once per 4 hours, whenever the specific activity exceeds a limit, as required by ACTION b.	1#, 2#, 3#, 4#
	b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION c.	1, 2
5. Isotopic Analysis of an Off-gas Sample Including Quantitative Measurements for at least Xe-133, Xe-135 and Kr-88	At least once per 31 days	1

---

*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

#Until the specific activity of the primary coolant system is restored to within its limits.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4.6.1-1 (1) curves A and A' for hydrostatic or leak testing; (2) curves B and B' for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curves C and C' for operations with a critical core other than low power PHYSICS TESTS, with:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 100°F in any one hour period,

c. A maximum temperature change of less than or equal to 20°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d. The reactor vessel flange and head flange metal temperature shall be maintained greater than or equal to 79°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figure 3.4.6.1-1 curves A and A', B and B', or C and C' as applicable, at least once per 30 minutes.

BEST COPY AVAILABLE

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 curves C and C' within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.

4.4.6.1.3 The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties, as required by 10 CFR 50, Appendix H in accordance with the schedule in Table 4.4.6.1.3-1. The results of these examinations shall be used to update the curves of Figure 3.4.6.1-1 based on the greater of the following criteria:

a. The actual shift in reference temperature for plate material from heat 5K3238-1 and weld metal 510-01205 as determined by Charpy impact test, or

b. The predicted shift in reference temperatures for plate material from heat 5K3025-1 as determined by Regulatory Guide 1.99, "Radiation Damage to Reactor Vessel Materials."

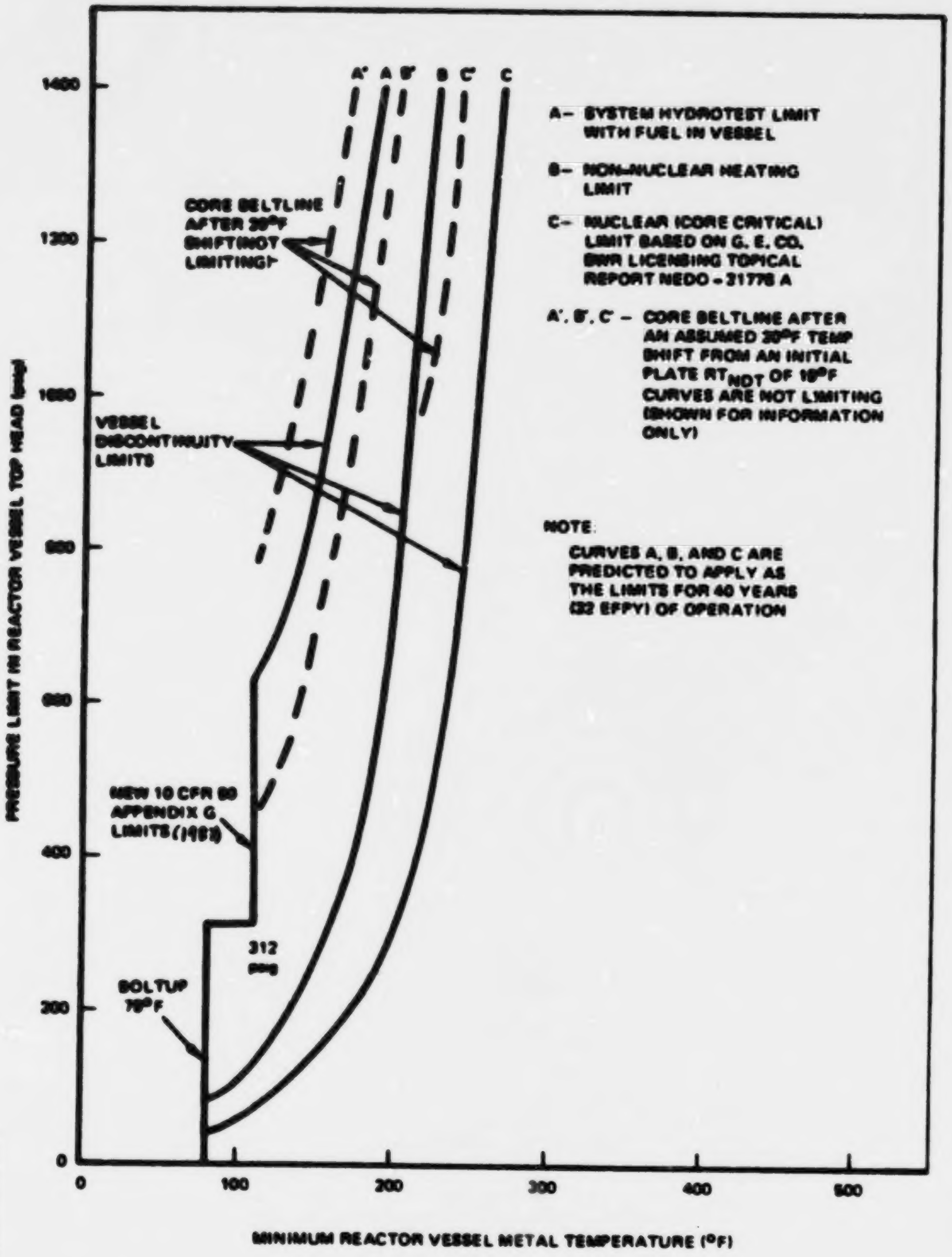
4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 70°F:

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

    1. $\leq$ 100°F, at least once per 12 hours.

    2. $\leq$ 80°F, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.



MINIMUM REACTOR PRESSURE VESSEL METAL TEMPERATURE VS. REACTOR VESSEL PRESSURE

Figure 3.4.6.1-1

BEST COPY AVAILABLE

TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR @ ¼ T	WITHDRAWAL TIME (EFPY)
1	30°	1.20	6
2	120°	1.20	15
3	300°	1.20	EOL

REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than 1020 psig.

APPLICABILITY: OPERATIONAL CONDITION 1* and 2*.

ACTION:

With the reactor steam dome pressure exceeding 1020 psig, reduce the pressure to less than 1020 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 The reactor steam dome pressure shall be verified to be less than 1020 psig at least once per 12 hours.

---

*Not applicable during anticipated transients.

BEST COPY AVAILABLE

199

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.7 Two main steam line isolation valves (MSIVs) per main steam line shall be OPERABLE with closing times greater than or equal to 3 and less than or equal to 5 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more MSIVs inoperable:

  1. Maintain at least one MSIV OPERABLE in each affected main steam line that is open and within 8 hours, either:

    a) Restore the inoperable valve(s) to OPERABLE status, or

    b) Isolate the affected main steam line by use of a deactivated MSIV in the closed position.

  2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.7 Each of the above required MSIVs shall be demonstrated OPERABLE by verifying full closure between 3 and 5 seconds when tested pursuant to Specification 4.0.5.

200

3/4.4.8 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.8 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.8.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.8 No requirements other than Specification 4.0.5.

3/4.4.9 RESIDUAL HEAT REMOVAL

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.9.1 Two[#] shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode loop shall be in operation[*],[##], with each loop consisting of:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 3, with reactor vessel pressure less than the RHR cut-in permissive setpoint.

ACTION:

a. With less than the above required RHR shutdown cooling mode loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible. Within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop. Be in at least COLD SHUTDOWN within 24 hours.[**]

b. With no RHR shutdown cooling mode loop or recirculation pump in operation, immediately initiate corrective action to return at least one loop to operation as soon as possible. Within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.1 At least one shutdown cooling mode loop of the residual heat removal system, one recirculation pump, or alternate method shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

[#] One RHR shutdown cooling mode loop may be inoperable for up to 2 hours for surveillance testing provided the other loop is OPERABLE and in operation or at least one recirculation pump is in operation.

[*] The shutdown cooling pump may be removed from operation for up to 2 hours per 8 hour period provided the other loop is OPERABLE.

[##] The RHR shutdown cooling mode loop may be removed from operation during hydrostatic testing.

[**] Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

BEST [illegible]

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.9.2 Two[#] shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode loop shall be in operation*,## with each loop consisting of:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 4 and heat losses to ambient** are not sufficient to maintain OPERATIONAL CONDITION 4.

ACTION:

a. With less than the above required RHR shutdown cooling mode loops OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop.

b. With no RHR shutdown cooling mode loop or recirculation pump in operation, within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.2 At least one shutdown cooling mode loop of the residual heat removal system, recirculation pump or alternate method shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

[#] One RHR shutdown cooling mode loop may be inoperable for up to 2 hours for surveillance testing provided the other loop is OPERABLE and in operation or at least one recirculation pump is in operation.

*The shutdown cooling pump may be removed from operation for up to 2 hours per 8 hour period provided the other loop is OPERABLE.

##The shutdown cooling mode loop may be removed from operation during hydrostatic testing.

**Ambient losses must be such that no increase in reactor vessel water temperature will occur (even though COLD SHUTDOWN conditions are being maintained).

203

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.1 The emergency core cooling systems shall be OPERABLE with:

a. The core spray system (CSS) consisting of two subsystems with each subsystem comprised of:

  1. Two OPERABLE core spray pumps, and

  2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

b. The low pressure coolant injection (LPCI) system of the residual heat removal system consisting of four subsystems with each subsystem comprised of:

  1. One OPERABLE LPCI pump, and

  2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

c. The high pressure coolant injection (HPCI) system consisting of:

  1. One OPERABLE HPCI pump, and

  2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

d. The automatic depressurization system (ADS) with five OPERABLE ADS valves.

APPLICABILITY: OPERATIONAL CONDITION 1, 2*, ** #, and 3*, **, ##.

---

[*] The HPCI system is not required to be OPERABLE when reactor steam dome pressure is less than or equal to 200 psig.

[**] The ADS is not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

[#] See Special Test Exception 3.10.6.

[##] Two LPCI subsystems of the RHR system may be inoperable in that they are aligned in the shutdown cooling mode when the reactor vessel pressure is less than the RHR shutdown cooling permissive setpoint.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. For the Core Spray system:

1. With one core spray subsystem inoperable, provided that at least two LPCI subsystem are OPERABLE, restore the inoperable core spray subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With both core spray subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. For the LPCI system:

1. With one LPCI subsystem inoperable, provided that at least one core spray subsystem is OPERABLE, restore the inoperable LPCI subsystem to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With two LPCI subsystems inoperable, provided that at least one core spray subsystem is operable, restore at least one LPCI subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With three LPCI subsystems inoperable, provided that both core spray subsystems are OPERABLE, restore at least two LPCI subsystems to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. With all four LPCI subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.*

c. For the HPCI system, provided the Core Spray System, the LPCI system, the ADS and the RCIC system are OPERABLE:

---

*Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

205

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

1. With the HPCI system inoperable, restore the HPCI system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤ 200 psig within the following 24 hours.

d. For the ADS:

1. With one of the above required ADS valves inoperable, provided the HPCI system, the core spray system and the LPCI system are OPERABLE, restore the inoperable ADS valve to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤ 100 psig within the next 24 hours.

2. With two or more of the above required ADS valves inoperable, be in at least HOT SHUTDOWN within 12 hours and reduce reactor steam dome pressure to ≤ 100 psig within the next 24 hours.

e. With a CSS and/or LPCI header ΔP instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 7 days or determine the ECCS header ΔP locally at least once per 12 hours; otherwise, declare the associated ECCS subsystem inoperable.

f. With a LPCI and/or CCS discharge line "keep filled" alarm instrumentation inoperable, perform Surveillance Requirement 4.5.1.a.1.a.

g. In the event an ECCS system is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

SURVEILLANCE REQUIREMENTS

4.5.1 The emergency core cooling systems shall be demonstrated OPERABLE by:

a. At least once per 31 days:

1. For the core spray system, the LPCI system, and the HPCI system:
    - a) Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.
    - b) Verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct* position.
2. For the HPCI system, verifying that the HPCI pump flow controller is in the correct position.

b. Verifying that, when tested pursuant to Specification 4.0.5:

1. The two core spray system pumps in each subsystem together develop a flow of at least 6350 gpm against a test line pressure corresponding to a reactor vessel pressure of $\geq$ 105 psi above suppression pool pressure.
2. Each LPCI pump in each subsystem develops a flow of at least 10,000 gpm against a test line pressure corresponding to a reactor vessel to primary containment differential pressure of $\geq$ 20 psid.
3. The HPCI pump develops a flow of at least 5600 gpm against a test line pressure corresponding to a reactor vessel pressure of 1000 psig when steam is being supplied to the turbine at 1000, +20, -80 psig.**

c. At least once per 18 months:

1. For the core spray system, the LPCI system, and the HPCI system, performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

---

*Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in position for another mode of operation.

**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

2. For the HPCI system, verifying that:
    a) The system develops a flow of at least 5600 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥ 200 psig, when steam is being supplied to the turbine at 200 + 15, -0 psig.**
    b) The suction is automatically transferred from the condensate storage tank to the suppression chamber on a condensate storage tank water level - low signal and on a suppression chamber - water level high signal.
3. Performing a CHANNEL CALIBRATION of the CSS, and LPCI system discharge line "keep filled" alarm instrumentation.
4. Performing a CHANNEL CALIBRATION of the CSS header ΔP instrumentation and verifying the setpoint to be ≤ the allowable value of 4.4 psid.#
5. Performing a CHANNEL CALIBRATION of the LPCI header ΔP instrumentation and verifying the setpoint to be ≤ the allowable value of 1.0 psid.#

d. For the ADS:

1. At least once per 31 days, performing a CHANNEL FUNCTIONAL TEST of the Primary Containment Instrument Gas System low-low pressure alarm system.
2. At least once per 18 months:
    a) Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.
    b) Manually opening each ADS valve when the reactor steam dome pressure is greater than or equal to 100 psig** and observing that either:
        1) The control valve or bypass valve position responds accordingly, or
        2) There is a corresponding change in the measured steam flow.
    c) Performing a CHANNEL CALIBRATION of the Primary Containment Instrument Gas System low-low pressure alarm system and verifying an alarm setpoint of 85 ± 2 psig on decreasing pressure.

**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

#Initial setpoint. Final setpoint to be determined during the startup test program.

BEST COPY AVAILABLE

208

3/4 5.2 ECCS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.5.2 At least two of the following shall be OPERABLE:

a. Core spray system subsystems with a subsystem comprised of:

1. Two OPERABLE core spray pumps, and

2. An OPERABLE flow path capable of taking suction from at least one of the following water sources and transferring the water through the spray sparger to the reactor vessel:

a) From the suppression chamber, or

b) When the suppression chamber water level is less than the limit or is drained, from the condensate storage tank containing at least 135,000 available gallons of water.

b. Low pressure coolant injection (LPCI) system subsystems each with a subsystem comprised of:

1. One OPERABLE LPCI pump, and

2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITION 4 and 5*.

ACTION:

a. With one of the above required subsystems inoperable, restore at least two subsystems to OPERABLE status within 4 hours or suspend all operations with a potential for draining the reactor vessel.

b. With both of the above required subsystems inoperable, suspend CORE ALTERATIONS and all operations with a potential for draining the reactor vessel. Restore at least one subsystem to OPERABLE status within 4 hours or establish SECONDARY CONTAINMENT INTEGRITY within the next 8 hours.

---

*The ECCS is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded, the spent fuel pool gates are removed, and water level is maintained within the limits of Specification 3.9.8 and 3.9.9.

SURVEILLANCE REQUIREMENTS

4.5.2.1 At least the above required ECCS shall be demonstrated OPERABLE per Surveillance Requirement 4.5.1.

4.5.2.2 The core spray system shall be determine OPERABLE at least once per 12 hours by verifying the condensate storage tank required volume when the condensate storage tank is required to be OPERABLE per Specification 3.5.2.a.2.b.

3/4.5.3 SUPPRESSION CHAMBER

LIMITING CONDITION FOR OPERATION

3.5.3 The suppression chamber shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2 and 3 with a contained water volume of at least 118,000 ft^3, equivalent to an indicated level of 74.5".

b. In OPERATIONAL CONDITION 4 and 5* with a contained volume of at least 57,390 ft^3, equivalent to an indicated level of 5.0" except that the suppression chamber level may be less than the limit or may be drained provided that:

1. No operations are performed that have a potential for draining the reactor vessel,

2. The reactor mode switch is locked in the Shutdown or Refuel position,

3. The condensate storage tank contains at least 135,000 available gallons of water, and

4. The core spray system is OPERABLE per Specification 3.5.2 with an OPERABLE flow path capable of taking suction from the condensate storage tank and transferring the water through the spray sparger to the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1, 2 or 3 with the suppression chamber water level less than the above limit, restore the water level to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4 or 5* with the suppression chamber water level less than the above limit or drained and the above required conditions not satisfied, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel and lock the reactor mode switch in the Shutdown position. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

---

*The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specification 3.9.8 and 3.9.9.

211

SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression chamber shall be determined OPERABLE by verifying the water level to be greater than or equal to:

a. 74.5" at least once per 24 hours in OPERATIONAL CONDITIONS 1, 2, and 3.

b. 5.0" at least once per 12 hours in OPERATIONAL CONDITIONS 4 and 5*.

4.5.3.2 With the suppression chamber level less than the above limit or drained in OPERATIONAL CONDITION 4 or 5*, at least once per 12 hours:

a. Verify the required conditions of Specification 3.5.3.b to be satisfied, or

b. Verify footnote conditions * to be satisfied.

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing the seals with gas at Pa, 48.1 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 La.

b. At least once per 31 days by verifying that all primary containment penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except as provided in Table 3.6.3-1 of Specification 3.6.3.

c. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. By verifying the suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

---

*See Special Test Exception 3.10.1

**Except valves, blind flanges, and deactivated automatic valves which are located inside the primary containment, and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been de-inerted since the last verification or more often than once per 92 days.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to L_a, 0.5 percent by weight of the containment air per 24 hours at P_a, 48.1 psig.

b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves*, valves which form the boundary for the long-term seal of the feedwater lines, and other valves which are hydrostatically tested per Table 3.6.3-1, subject to Type B and C tests when pressurized to P_a, 48.1 psig.

c. *Less than or equal to 46.0 scfh combined through all four main steam lines when tested at 5 psig (seal system ΔP).

d. A combined leakage rate of less than or equal to 10 gpm for all containment isolation valves which form the boundary for the long-term seal of the feedwater lines in Table 3.6.3-1, when tested at 1.10 Pa, 52.9 psig.

e. A combined leakage rate of less than or equal to 10 gpm for all other containment isolation valves in hydrostatically tested lines in Table 3.6.3-1 which penetrate the primary containment, when tested at Pa, 48.1 psig Δp.

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

ACTION:

With:

a. The measured overall integrated primary containment leakage rate exceeding 0.75 L_a or

b. The measured combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves*, valves which form the boundary for the long-term seal of the feedwater lines, and other valves which are hydrostatically tested per Table 3.6.3-1, subject to Type B and C tests exceeding 0.60 L_a, or

c. The measured leakage rate exceeding 46.0 scfh combined through all four main steam lines, or

d. The measured combined leakage rate for all containment isolation valves which form the boundary for the long-term seal of the feedwater lines in Table 3.6.3-1 exceeding 10 gpm, or

e. The measured combined leakage rate for all other containment isolation valves in hydrostatically tested lines in Table 3.6.3-1 which penetrate the primary containment exceeding 10 gpm,

restore:

a. The overall integrated leakage rate(s) to less than or equal to 0.75 L_a, and

---

*Exemption to Appendix "J" of 10 CFR 50.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

b. The combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves*, valves which form the boundary for the long-term seal of the feedwater lines, and other valves which are hydrostatically tested per Table 3.6.3-1, subject to Type B and C tests to less than or equal to 0.60 L_a, and

c. The leakage rate to less than or equal to 46.0 scfh combined through all four main steam lines, and

d. The combined leakage rate for all containment isolation valves which form the boundary for the long-term seal of the feedwater lines in Table 3.6.3-1 to less than or equal to 10 gpm, and

e. The combined leakage rate for all other containment isolation valves in hydrostatically tested lines in Table 3.6.3-1 which penetrate the primary containment to less than or equal to 10 gpm,

prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4 - 1972:

a. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 ± 10 month intervals during shutdown at P_a, 48.1 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

b. If any periodic Type A test fails to meet 0.75 L_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a, at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

   1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 0.25 L_a.
   2. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.
   3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be between 0.75 L_a and 1.25 L_a.

---

*Exemption to Appendix "J" of 10 CFR 50.

215

SURVEILLANCE REQUIREMENTS (Continued)

The formula to be used is: $[L_o + L_{am} - 0.25\ L_a] \le L_c \le [L_o + L_{am} + 0.25\ L_a]$ where $L_c \equiv$ supplement test result; $L_o \equiv$ superimposed leakage; and $L_a \equiv$ measured Type A leakage.

d. Type B and C tests shall be conducted with gas at P_a, 48.1 psig*, at intervals no greater than 24 months except for tests involving:

1. Air locks,
2. Main steam line isolation valves,
3. Valves pressurized with fluid from a seal system,
4. All containment isolation valves in hydrostatically tested lines in Table 3.6.3-1 which penetrate the primary containment, and
5. Purge supply and exhaust isolation valves with resilient material seals.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

f. Main steam line isolation valves shall be leak tested at least once per 18 months.

g. Containment isolation valves which form the boundary for the long-term seal of the feedwater lines in Table 3.6.3-1 shall be hydrostatically tested at 1.10 P_a, 52.9 psig, at least once per 18 months.

h. All containment isolation valves in hydrostatically tested lines in Table 3.6.3-1 which penetrate the primary containment shall be leak tested at least once per 18 months.

i. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirements 4.6.1.8.2 and 4.6.1.8.3.

j. The provisions of Specification 4.0.2 are not applicable to Specifications 4.6.1.2.a, 4.6.1.2.b, 4.6.1.2.c, 4.6.1.2.d, and 4.6.1.2.e.

---

*Unless a hydrostatic test is required per Table 3.6.3-1.

PRIMARY CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each primary containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, 48.1 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

a. With one primary containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. The provisions of Specification 3.0.4 are not applicable.

b. With the primary containment air lock inoperable, except as a result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

---

*See Special Test Exception 3.10.1.

4.6.1.3 Each primary containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage rate less than or equal to 5 scf per hour when the gap between the door seals is pressurized to 10.0 psig.

b. By conducting an overall air lock leakage test at P_a, 48.1 psig, and by verifying that the overall air lock leakage rate is within its limit:

   1. At least once per 6 months[#], and

   2. Prior to establishing PRIMARY CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the airlock sealing capability*, and

   3. Prior to establishing PRIMARY CONTAINMENT INTEGRITY when maintenance has not been performed on the air lock that could affect the air lock sealing capability, a seal test may be performed in lieu of the overall air lock leakage test. The acceptance criteria and test pressure shall be as specified in 4.6.1.3.a.

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.**

[#] The provisions of Specification 4.0.2 are not applicable.

*Exemption to Appendix J of 10 CFR 50.

**Except that the inner door need not be opened to verify interlock OPERABILITY when the primary containment is inerted, provided that the inner door interlock is tested within 8 hours after the primary containment has been de-inerted.

MSIV SEALING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.4 Two independent MSIV sealing system (MSIVSS) subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With one MSIV sealing system subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 Each MSIV sealing system subsystem shall be demonstrated OPERABLE:

a. At least once per 92 days by cycling each testable motor-operated valve except the Main Steam Stop Valves (MSSVs) through at least one complete cycle of full travel.

b. During each COLD SHUTDOWN, if not performed within the previous 92 days, by cycling each motor-operated valve including the Main Steam Stop Valves (MSSVs) not testable during operation through a least one complete cycle of full travel.

c. At least once per 18 months by performance of a functional test of the subsystem throughout its operating sequence, and verifying that each interlock and timer operates as designed and each automatic valve actuates to its correct position.

d. By verifying the control instrumentation to be OPERABLE by performance of a:

  1. CHANNEL CHECK at least once per 24 hours,
  2. CHANNEL FUNCTIONAL TEST at least once per 92 days, and
  3. CHANNEL CALIBRATION at least once per 18 months.

PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.5 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.5.1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the structural integrity of the primary containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5.1 The structural integrity of the exposed accessible interior and exterior surfaces of the primary containment shall be determined during the shutdown for each Type A containment leakage rate test by a visual inspection of those surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.5.2 Reports Any abnormal degradation of the primary containment structure detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.2 within 30 days. This report shall include a description of the condition of the containment, the inspection procedure, and the corrective actions taken.

DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.6 Drywell and suppression chamber internal pressure shall be maintained between -0.5 and +1.5 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the drywell and/or suppression chamber internal pressure outside of the specified limits, restore the internal pressure to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The drywell and suppression chamber internal pressure shall be determined to be within the limits at least once per 12 hours.

DRYWELL AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.7 Drywell average air temperature shall not exceed 135°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the drywell average air temperature greater than 135°F, reduce the average air temperature to within the limit within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The drywell average air temperature shall be the volumetric average of the temperatures at the following locations and shall be determined to be within the limit at least once per 24 hours:

	Elevation Zone	Approximate Azimuth*
a.	86'11"-112'8" (under vessel)	90°, 225°, 90°, 270°
b.	86'11"-111'10" (outside of pedestal)	135°, 300°, 100°, 190°
c.	111'10"-139'2"	55°, 240°, 155°, 315°
d.	139'2"-168'0"	45°, 215°, 0°, 90°, 180°, 270°
e.	168'0"-192'7"	95°, 130°, 300°, 355°, 45°, 225°

---

*At least one reading from each elevation zone is required for a volumetric average calculation.

DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.8 The 26-inch drywell purge supply and exhaust isolation valves and the 24-inch suppression chamber purge supply and exhaust isolation valves, and the 6-inch nitrogen supply valve shall be OPERABLE and sealed closed.*

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With a 26-inch drywell purge supply or exhaust isolation valve, or a 24-inch suppression chamber purge supply or exhaust isolation valve or the 6-inch nitrogen supply valve open or not sealed closed,* close or seal the valves(s) or otherwise isolate the penetration within four hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With a drywell purge supply or exhaust isolation valve, or a suppression chamber purge supply or exhaust isolation valve or the nitrogen supply valve, with resilient material seals having a measured leakage rate exceeding the limit of Surveillance Requirements 4.6.1.8.2 and/or 4.6.1.8.3, restore the inoperable valve(s) to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.8.1 Each 26-inch drywell purge supply and exhaust isolation valve, and 24-inch suppression chamber purge supply and exhaust isolation valve and the 6-inch nitrogen supply valve, shall be verified to be sealed closed* at least once per 31 days.

4.6.1.8.2 At least once per 6 months on a STAGGERED TEST BASIS each sealed closed 26-inch drywell purge supply and exhaust isolation valve, and 24-inch suppression chamber purge supply and exhaust isolation valve, and the 6-inch nitrogen supply valve, with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.05 L_a per penetration when pressurized to P_a 48.1 psig.

4.6.1.8.3 At least once per 92 days the 26-inch drywell purge inboard exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.05 L_a per penetration when pressurized to P_a 48.1 psig.

---

*The 26-inch drywell purge inboard exhaust valve is not required to be sealed closed and may be opened in series with the 2 inch vent line bypass valve during periods of containment pressure control.

3/4.6.2 DEPRESSURIZATION SYSTEMS

SUPPRESSION CHAMBER

LIMITING CONDITION FOR OPERATION

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. Volume between 118,000 ft^3 and 122,000 ft^3, equivalent to an indicated level between 74.5" and 78.5" and a

2. Maximum average temperature of 95°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

a) 105°F during testing which adds heat to the suppression chamber.

b) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

3. Maximum average temperature of 95°F during OPERATIONAL CONDITION 3, except that the maximum average temperature may be permitted to increase to 120°F with the main steam line isolation valves closed following a scram.

b. A total leakage between the suppression chamber and drywell of less than the equivalent leakage through a 1-inch diameter orifice at a differential pressure of 0.80 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the suppression chamber average water temperature greater than 95°F and THERMAL POWER greater than 1% of RATED THERMAL POWER, restore the average temperature to less than or equal to 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

1. With the suppression chamber average water temperature greater than 105°F during testing which adds heat to the suppression chamber, stop all testing which adds heat to the suppression chamber and restore the average temperature to less than 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With the suppression chamber average water temperature greater than 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

224

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

3. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

c. With one suppression pool water temperature monitoring channel inoperable, restore the inoperable channel(s) to OPERABLE status within 7 days or verify suppression pool temperature to be within the limits at least once per 12 hours.

d. With both suppression pool water temperature monitoring channels inoperable, restore at least one inoperable channel to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With one suppression chamber water level instrumentation channel inoperable, restore the inoperable narrow range suppression chamber water level channel to OPERABLE status within 7 days or verify suppression pool water level to be within the limits at least once per 12 hours.

f. With both suppression chamber water level instrumentation channels inoperable, restore at least one narrow range suppression chamber water level channel to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

g. With the drywell-to-suppression chamber bypass leakage in excess of the limit, restore the bypass leakage to within the limit prior to increasing reactor coolant temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

a. By verifying the suppression chamber water volume to be within the limits at least once per 24 hours.

b. At least once per 24 hours in OPERATIONAL CONDITION 1 or 2 by verifying the suppression chamber average water temperature to be less than or equal to 95°F, except:

1. At least once per 5 minutes during testing which adds heat to the suppression chamber, by verifying the suppression chamber average water temperature less than or equal to 105°F.

2. At least once per hour when suppression chamber average water temperature is greater than 95°F, by verifying:

    a) Suppression chamber average water temperature to be less than or equal to 110°F, and

225

b) THERMAL POWER to be less than or equal to 1% of RATED THERMAL POWER.

c) At least once per 30 minutes in OPERATIONAL CONDITION 3 following a scram with suppression chamber average water temperature greater than 95°F, by verifying suppression chamber average water temperature less than or equal to 120°F.

c. By an external visual examination of the suppression chamber after safety/relief valve operation with the suppression chamber average water temperature greater than or equal to 177°F and reactor coolant system pressure greater than 100 psig.

d. At least once per 18 months by a visual inspection of the accessible interior and exterior of the suppression chamber.

e. By verifying all temperature elements used by the suppression pool water temperature monitoring system OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,
2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and
3. CHANNEL CALIBRATION at least once per 18 months,

with the water high temperature alarm setpoint for:

1. First setpoint ≤ 95°F
2. Second setpoint ≤ 105°F
3. Third setpoint ≤ 110°F
4. Fourth setpoint ≤ 120°F

f. By verifying both suppression chamber water level instrumentation channels OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,
2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and
3. CHANNEL CALIBRATION at least once per 18 months,

With the water level alarm setpoint for:

1. High water level ≤ 78.5"
2. Low water level ≥ 65.0"

g. At least once per 18 months by conducting a drywell-to-suppression chamber bypass leak test at an initial differential pressure of 0.80 psi and verifying that the differential pressure does not decrease by more than 0.24 inch of water per minute for a period of 10 minutes. If any drywell-to-suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 9 months until two consecutive tests meet the specified limit, at which time the 18 month test schedule may be resumed.

SUPPRESSION POOL SPRAY

LIMITING CONDITION FOR OPERATION

3.6.2.2 The suppression pool spray mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

a. One OPERABLE RHR pump, and

b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHR heat exchanger and the suppression pool spray sparger.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one suppression pool spray loop inoperable, restore the inoperable loop to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both suppression pool spray loops inoperable, restore at least one loop to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN* within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The suppression pool spray mode of the RHR system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. By verifying that each of the required RHR pumps develops a flow of at least 500 gpm on recirculation flow through the RHR heat exchanger and suppression pool spray sparger when tested pursuant to Specification 4.0.5.

---

*Whenever both RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

SUPPRESSION POOL COOLING

LIMITING CONDITION FOR OPERATION

3.6.2.3 The suppression pool cooling mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

a. One OPERABLE RHR pump, and

b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one suppression pool cooling loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both suppression pool cooling loops inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN* within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 The suppression pool cooling mode of the RHR system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. By verifying that each of the required RHR pumps develops a flow of at least 10,000 gpm on recirculation flow through the RHR heat exchanger and the suppression pool when tested pursuant to Specification 4.0.5.

---

*Whenever both RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The primary containment isolation valves and the reactor instrumentation line excess flow check valves shown in Table 3.6.3-1 shall be OPERABLE with isolation times less than or equal to those shown in Tabl- 3.6.3-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more of the primary containment isolation valves shown in Table 3.6.3-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours either:

   1. Restore the inoperable valve(s) to OPERABLE status, or

   2. Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolated position,* or

   3. Isolate each affected penetration by use of at least one closed manual valve or blind flange.*

   4. The provisions of Specification 3.0.4 are not applicable provided that within 4 hours the affected penetration is isolated in accordance with ACTION a.2. or a.3. above, and provided that the associated system, if applicable, is declared inoperable and the appropriate ACTION statements for that system are performed.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one or more of the reactor instrumentation line excess flow check valves shown in Table 3.6.3-1 inoperable, operation may continue and the provisions of Specifications 3.0.3 and 3.0.4 are not applicable provided that within 4 hours either:

   1. The inoperable valve is returned to OPERABLE status, or

   2. The instrument line is isolated and the associated instrument is declared inoperable.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

---

*Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative control.

4.6.3.1 Each primary containment isolation valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

4.6.3.2 Each primary containment automatic isolation valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each primary containment power operated or automatic valve shown in Table 3.6.3-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 Each reactor instrumentation line excess flow check valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve checks flow.

4.6.3.5 Each traversing in-core probe system explosive isolation valve shall be demonstrated OPERABLE*:

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 18 months by removing the explosive squib from at least one explosive valve such that each explosive squib in each explosive valve will be tested at least once per 90 months, and initiating the explosive squib. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life or operating life, as applicable.

---

*Exemption to Appendix J of 10 CFR Part 50.

TABLE 3.6.3-1

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
A. Automatic Isolation Valves				
1. Group 1 - Main Steam system				
(a) Main Steam Isolation Valves (MSIVs)				M-41-1
Inside:				
Line A HV-F022A (AB-V028)	P1A	5	1	
Line B HV-F022B (AB-V029)	P1B	5	1	
Line C HV-F022C (AB-V030)	P1C	5	1	
Line D HV-F022D (AB-V031)	P1D	5	1	
Outside:				
Line A HV-F028A (AB-V032)	P1A	5	1	
Line B HV-F028B (AB-V033)	P1B	5	1	
Line C HV-F028C (AB-V034)	P1C	5	1	
Line D HV-F028D (AB-V035)	P1D	5	1	
(b) Main Steam Line Drain Isolation				M-41-1
Inside: HV-F016 (AB-V039)	P12	30	3	
Outside:				
Line A HV-F067A (AB-V059)	P1A	45	1	
Line B HV-F067B (AB-V060)	P1B	45	1	
Line C HV-F067C (AB-V061)	P1C	45	1	
Line D HV-F067D (AB-V062)	P1D	45	1	
HV-F019 (AB-V040)	P12	30	3	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
(c) MSIV Sealing System Isolation Valves				M-72-1
Outside:				
Line A HV-5834A (KP-V010)	P1A	45	1	
Line B HV-5835A (KP-V009)	P1B	45	1	
Line C HV-5836A (KP-V008)	P1C	45	1	
Line D HV-5837A (KP-V007)	P1D	45	1	
2. Group 2 - Reactor Recirculation Water Sample System				
(a) Reactor Recirculation Water Sample Line Isolation Valves				M-43-1
Inside: BB-SV-4310	P17	15	3	
Outside: BB-SV-4311	P17	15	3	
3. Group 3 - Residual Heat Removal (RHR) System				
(a) RHR Suppression Pool Cooling Water & System Test Isolation Valves				M-51-1
Outside:				
Loop A: HV-F024A (BC-V124)	P212B	180	4	
HV-F010A (BC-V125)	P212B	180	4	
Outside:				
Loop B: HV-F024B (BC-V028)	P212A	180	4	
HV-F010B (BC-V027)	P212A	180	4	
(b) RHR to Suppression Chamber Spray Header Isolation Valves				M-51-1
Outside:				
Loop A: HV-F027A (BC-V112)	P214B	75	3	
Loop B: HV-F027B (BC-V015)	P214A	75	3	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
(c) RHR Shutdown Cooling Suction Isolation Valves				M-51-1
Inside: HV-F009 (BC-V071)	P3	45	3	
Outside: HV-F008 (BC-V164)	P3	45	3	
(d) RHR Head Spray Isolation Valves				M-51-1
Inside: HV-F022 (BC-V021)	P10	60	3	
Outside: HV-F023 (BC-V020)	P10	60	3	
(e) RHR Shutdown Cooling Return Isolation Valves				M-51-1
Outside:				
Loop A: HV-F015A (BC-V110)	P4B	45	3	
Loop B: HV-F015B (BC-V013)	P4A	45	3	
4. Group 4 - Core Spray System				
Outside:				
(a) Core Spray Test to Suppression Pool Isolation Valves				M-52-1
Loop A: HV-F015A (BE-V025)	P217B	80	4	
Loop B: HV-F015B (BE-V026)	P217A	80	4	
5. Group 5 - High Pressure Coolant Injection (HPCI) System				
(a) HPCI Turbine Steam Supply Isolation Valves				M-55-1
Inside: HV-F002 (FD-V001)	P7	NA	3	
HV-F100 (FD-V051)	P7	NA	3	
Outside: HV-F003 (FD-V002)	P7	NA	3	
(b) HPCI Pump Suction Isolation Valve				M-55-1
Outside:				
HV-F042 (BJ-V009)	P202	NA	4	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
(c) HPCI Turbine Exhaust Isolation Valve to Vacuum Breaker Network				M-55-1
Outside: HV-F075 (FD-V007)	P201/P204	NA	4	
(d) HPCI and RCIC Vacuum Network Isolation Valve				M-55-1
Outside: HV-F079 (FD-V010)	P204/P201	NA	3	
6. Group 6 - Reactor Core Isolation Cooling (RCIC) System				
(a) RCIC Turbine Steam Supply Isolation Valves				M-49-1
Inside: HV-F007 (FC-V001)	P11	NA	3	
HV-F076 (FC-V048)	P11	NA	3	
Outside: HV-F008 (FC-V002)	P11	NA	3	
(b) RCIC Turbine Exhaust Isolation Valve to Vacuum Breaker Network				M-49-1
Outside: HV-F062 (FC-V006)	P207/P204	NA	4	
(c) HPCI and RCIC Vacuum Network Isolation Valve				M-49-1
Outside: HV-F084 (FC-V007)	P204/P207	NA	3	
7. Group 7 - Reactor Water Cleanup (RWCU) System				
(a) RWCU Supply Isolation Valves				M-44-1
Inside: HV-F001 (BG-V001)	P9	45	3	
Outside: HV-F004 (BG-V002)	P9	45	3	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
8. Group 8 - Torus Water Cleanup (TWC) System				
(a) TWC Suction Isolation Valves				M-53-1
Outside:				
HV-4680 (EE-V003)	P223	45	4	
HV-4681 (EE-V004)	P223	45	4	
(b) TWC Return Isolation Valves				M-53-1
Outside:				
HV-4652 (EE-V002)	P222	45	4	
HV-4679 (EE-V001)	P222	45	4	
9. Group 9 - Drywell Sumps				
(a) Drywell Floor Drain Sump Discharge Isolation Valves				M-61-1
Inside: HV-F003 (HB-V005)	P25	30	3	
Outside: HV-F004 (HB-V006)	P25	30	3	
(b) Drywell Equipment Drain Sump Discharge Isolation Valves				M-61-1
Inside: HV-F019 (HB-V045)	P26	30	3	
Outside: HV-F020 (HB-V046)	P26	30	3	
10. Group 10 - Drywell Coolers				
(a) Chilled Water to Drywell Coolers Isolation Valves				M-87-1
Inside:				
Loop A: HV-9531B1 (GB-V081)	P8B	60	3	
Loop B: HV-9531B3 (GB-V083)	P38A	60	3	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
Outside:				
Loop A: HV-9531A1 (GB-V048)	P8B	60	3	
Loop B: HV-9531A3 (GB-V070)	P38A	60	3	
(b) Chilled Water from Drywell Coolers Isolation Valves				M-87-1
Inside:				
Loop A: HV-9531B2 (GB-V082)	P8A	60	3	
Loop B: HV-9531B4 (GB-V084)	P38B	60	3	
Outside:				
Loop A: HV-9531A2 (GB-V046)	P8A	60	3	
Loop B: HV-9531A4 (GB-V071)	P38B	60	3	
11. Group 11 - Recirculation Pump System				
(a) Recirculation Pump Seal Water Isolation Valves				M-43-1
Outside:				
Loop A: HV-3800A (BF-V098)	P19	45	3	
Loop B: HV-3800B (BF-V099)	P20	45	3	
12. Group 12 - Containment Atmosphere Control System				
(a) Drywell Purge Supply Isolation Valves				M-57-1
Outside:				
HV-4956 (GS-V009)	P22	15	3, 8	
HV-4979 (GS-V021)	P22/220	15	3, 8	
(b) Drywell Purge Exhaust Isolation Valves				M-57-1
Outside:				
HV-4951 (GS-V025)	P23	15	3	
HV-4950 (GS-V026)	P23	15	3, 8	
HV-4952 (GS-V024)	P23	15	3, 8	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
(c) Suppression Chamber Purge Supply Isolation Valves				M-57-1
Outside:				
HV-4980 (GS-V020)	P22/P220	15	3, 8	
HV-4958 (GS-V022)	P220	15	3, 8	
(d) Suppression Chamber Purge Exhaust Isolation Valves				M-57-1
Outside:				
HV-4963 (GS-V076)	P219	15	3	
HV-4962 (GS-V027)	P219	15	3, 8	
HV-4964 (GS-V028)	P219	15	3, 8	
(e) Nitrogen Purge Isolation Valves				M-57-1
Outside:				
HV-4974 (GS-V053)	J7D/J202	45	3	
HV-4978 (GS-V023)	P22/P220	15	3, 8	
13. Group 13 - Hydrogen/Oxygen (H2/02) Analyzer System				
(a) Drywell H2/02 Analyzer Inlet Isolation Valves				M-57-1
Outside:				
Loop A: HV-4955A (GS-V045)	J9E	45	3	
HV-4983A (GS-V046)	J9E	45	3	
HV-4984A (GS-V048)	J10C	45	3	
HV-5019A (GS-V047)	J10C	45	3	
Outside:				
Loop B: HV-4955B (GS-V031)	J3B	45	3	
HV-4983B (GS-V032)	J3B	45	3	
HV-4984B (GS-V034)	J7D/J202	45	3	
HV-5019B (GS-V033)	J7D	45	3	

237

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
(b) Suppression Chamber H2/02 Analyzer Inlet Isolation Valves				M-57-1
Outside:				
Loop A: HV-4965A (GS-V050)	J212	45	3	
HV-4959A (GS-V049)	J212	45	3	
Outside:				
Loop B: HV-4965B (GS-V041)	J210	45	3	
HV-4959B (GS-V040)	J210	45	3	
(c) H2/02 Analyzer Return to Suppression Chamber Isolation Valves				M-57-1
Outside:				
Loop A: HV-4966A (GS-V051)	J201	45	3	
HV-5022A (GS-V052)	J201	45	3	
Outside:				
Loop B: HV-4966B (GS-V042)	J202	45	3	
HV-5022B (GS-V043)	J202/J7D	45	3	
14. Group 14 - Containment Hydrogen Recombination (CHR) System				
(a) CHR Supply Isolation Valves				M-58-1
Outside:				
Loop A: HV-5050A (GS-V002)	P23	45	3	
HV-5052A (GS-V003)	P23	45	3	
Outside:				
Loop B: HV-5050B (GS-V004)	P22	45	3	
HV-5052B (GS-V005)	P22	45	3	
(b) CHR Return Isolation Valves				M-58-1
Outside:				
Loop A: HV-5053A (GS-V008)	P220	45	3	
HV-5054A (GS-V010)	P220	45	3	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
Outside:				
Loop B: HV-5053B (GS-V006)	P219	45	3	M-58-1
HV-5054B (GS-V007)	P219	45	3	
15. Group 15 - Primary Containment Instrument Gas System (PCIGS)				
(a) PCIGS Drywell Supply Header Isolation Valves				M-59-1
Inside:				
Loop A: HV-5152A (KL-V028)	P28B	45	3	
Loop B: HV-5152B (KL-V026)	P28A	45	3	
Outside:				
Loop A: HV-5126A (KL-V027)	P28B	45	3	
Loop B: HV-5126B (KL-V025)	P28A	45	3	
(b) PCIGS Drywell Suction Isolation Valves				M-59-1
Inside:				
HV-5148 (KL-V001)	P39	45	3	
Outside:				
Loop A: HV-5147 (KL-V002)	P39	45	3	
Loop B: HV-5162 (KL-V049)	P39	45	3	
(c) PCIGS Suppression Chamber Supply Isolation Valves				M-59-1
Outside:				
HV-5154 (KL-V018)	J211	15	3	
HV-5155 (KL-V019)	J211	15	3	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
16. Group 16 - Reactor Auxiliaries Cooling System (RACS)				
(a) RACS Supply Isolation Valves				M-13-1
Inside: HV-2554 (ED-V020)	P29	45	3	
Outside: HV-2553 (ED-V019)	P29	45	3	
(b) RACS Return Isolation Valves				M-13-1
Inside: HV-2556 (ED-V022)	P30	45	3	
Outside: HV-2555 (ED-V021)	P30	45	3	
17. Group 17 - Traversing In-core Probe (TIP) System				
(a) TIP Probe Guide Tube Isolation Valves				M-59-1
Outside:				
SV-J004A-1 (SE-V026)	P34A	15	3	
SV-J004A-2 (SE-V027)	P34B	15	3	
SV-J004A-3 (SE-V028)	P34C	15	3	
SV-J004A-4 (SE-V029)	P34D	15	3	
SV-J004A-5 (SE-V030)	P34E	15	3	
(b) TIP Purge System Isolation Valve				M-59-1
Outside:				
HV-5161 (SE-V004)	P34F	15	3	
18. Group 18 - Reactor Coolant Pressure Boundary (RCPB) Leakage Detection System				
(a) Drywell Leak Detection Radiation Monitoring System (DLD-RMS) Inlet Isolation Valves				M-25-1
Outside:				
HV-5018 (SK-V005)	J8C	45	3	
HV-4953 (SK-V006)	J8C	45	3	

240

TABLE 3.6.3-1 (Continued)
PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
(b) DLD-RMS Return Isolation Valves				M-25-1
Outside:				
HV-4957 (SK-V008)	J5A	45	3	
HV-4981 (SK-V009)	J5A	45	3	
B. Remote Manual Isolation Valves				
1. Group 21 - Feedwater System				
(a) Feedwater Isolation Valves				M-41-1
Outside Check Valves				
HV-F032B (AE-V001)	P2A	NA	2	
HV-F032A (AE-V005)	P2B	NA	2	
(b) Reactor Water Cleanup System Return				
Outside:				
HV-F039 (AE-V021)	P2A&B	NA	2	M-44-1
2. Group 22 - High Pressure Coolant Injection (HPCI) System				
(a) Core Spray Discharge Valve				
Outside:				
HV-F006 (BJ-V001)	P5B	NA	3	M-55-1
(b) Turbine Exhaust Valve				
Outside:				
HV-F071 (FD-V006)	P201	NA	4	M-55-1
(c) HPCI Minimum Return Line Valve				
Outside:				
HV-F012 (BJ-V016)	P203	NA	4	M-55-1
(d) Feedwater Line Discharge Valve				
Outside:				
HV-8278 (BJ-V059)	P2B	NA	2	M-55-1
3. Group 23 - Reactor Core Isolation Cooling (RCIC) System				
(a) RCIC Turbine Exhaust Valve				
Outside:				
HV-F059 (FC-V005)	P207	NA	4	M-49-1

241

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
Outside: (b) RCIC Pump Suction Isolation Valve HV-F031 (BD-V003)	P208	NA	4	M-49-1
Outside: (c) RCIC Minimum Return Line Isolation Valve SV-F019	P209	NA	4	M-49-1
Outside: (d) RCIC Vacuum Pump Discharge HV-F060 (FC-V011)	P210	NA	4	M-49-1
(e) Feedwater Line Discharge Valve Outside: HV-F013 (BD-V005)	P2A	NA	2	M-49-1
4. Group 25 - Core Spray System				
(a) Core Spray injection Valves Outside:				M-52-1
Loop A&C HV-F005A (BE-V007)	P5B	NA	3	
Loop B&D HV-F005B (BE-V003)	P5A	NA	3	
(b) Core Spray Suppression Pool Suction Valves Outside:				M-52-1
Loop A HV-F001A (BE-V017)	P216D	NA	4	
Loop B HV-F001B (BE-V019)	P216A	NA	4	
Loop C HV-F001C (BE-V018)	P216C	NA	4	
Loop D HV-F001D (BE-V020)	P216B	NA	4	
(c) Core Spray Minimum Flow Valves Outside:				M-52-1
Loop A&C HV-F031A (BE-V035)	P217B	NA	4	
Loop B&D HV-F031B (BE-V036)	P217A	NA	4	
(d) Core Spray Injection Line Bypass Valves Inside:				M-52-1
HV-F039A (BE-V071)	P5B	NA	3	
HV-F039B (BE-V072)	P5A	NA	3	

242

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
5. Group 26 - Residual Heat Removal System				
(a) Low Pressure Coolant Injection Valves				M-51-1
Outside:				
Loop A: HV-F017A (BC-V113)	P6C	NA	3	
Loop B: HV-F017B (BC-V016)	P6B	NA	3	
Loop C: HV-F017C (BC-V101)	P6D	NA	3	
Loop D: HV-F017D (BC-V004)	P6A	NA	3	
(b) RHR Containment Spray				M-51-1
Outside:				
Loop A: HV-F021A (BC-V116)	P24B	NA	3	
HV-F016A (BC-V115)	P24B	NA	3	
Loop B: HV-F021B (BC-V019)	P24A	NA	3	
HV-F016B (BC-V018)	P24A	NA	3	
(c) RHR Suppression Pool Suction				M-51-1
Outside:				
Loop A: HV-F004A (BC-V103)	P211C	NA	4	
Loop B: HV-F004B (BC-V006)	P211B	NA	4	
Loop C: HV-F004C (BC-V098)	P211D	NA	4	
Loop D: HV-F004D (BC-V001)	P211A	NA	4	
(d) RHR Minimum Flow Isolation Valves				M-51-1
Outside:				
Loop A: HV-F007A (BC-V128)	P212B	NA	4	
Loop B: HV-F007B (BC-V031)	P212A	NA	4	
Loop C: HV-F007C (BC-V131)	P212B	NA	4	
Loop D: HV-F007D (BC-V034)	P212A	NA	4	

243

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
(e) Bypass Valves on LPCI Injection Lines				
Inside:				M-51-1
HV-F146A (BC-V119)	P6C	NA	3	
HV-F146B (BC-V120)	P6B	NA	3	
HV-F146C (BC-V121)	P6D	NA	3	
HV-F146D (BC-V122)	P6A	NA	3	
(f) Bypass Valves on Shutdown Cooling Return Lines				
Inside:				M-51-1
HV-F122A (BC-V117)	P4B	NA	3	
HV-F122B (BC-V118)	P4A	NA	3	
6. Group 27 - Standby Liquid Control				
Outside:				M-48-1
HV-F006A (BH-V028)	P18	NA	3	
HV-F006B (BH-V054)	P18	NA	3	
7. Group 28 - Containment Atmosphere Control System				
Supression Chamber Vacuum Relief				
Outside:				M-57-1
HV-5031 (GS-V038)	P220	NA	3	
HV-5029 (GS-V080)	P219	NA	3	
8. Group 69 - TIP System				
Explosive Shear Valves				
Outside:				M-59-1
SE-XV-J004B1 (SE-V021)	P34A	NA	7	
SE-XV-J004B2 (SE-V022)	P34B	NA	7	
SE-XV-J004B3 (SE-V023)	P34C	NA	7	
SE-XV-J004B4 (SE-V024)	P34D	NA	7	
SE-XV-J004B5 (SE-V025)	P34E	NA	7	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
9. Group 29 - HPCI System				
Suppression Pool Level Instrumentation Isolation				M-55-1
Outside:				
HV-4803 (BJ-V500)	J209	NA	6	
HV-4804 (BJ-V501)	P228	NA	6	
HV-4865 (BJ-V502)	J217	NA	6	
HV-4866 (BJ-V503)	J219	NA	6	
10. Group 30 - Post-Accident Sampling System				
Liquid Sampling				M-38-0
Outside:				
RC-SV-0643A	P227	NA	3	
RC-SV-0643B	P227	NA	3	
RC-SV-8903A	J50	NA	3	
RC-SV-8903B	J50	NA	3	
Gas Sampling				M-38-0
Outside:				
RC-SV-0730A	J7E	NA	3	
RC-SV-0730B	J7E	NA	3	
RC-SV-0731A	J10E	NA	3	
RC-SV-0731B	J10E	NA	3	
RC-SV-0728A	J206	NA	3	
RC-SV-0728B	J206	NA	3	
RC-SV-0729A	J221	NA	3	
RC-SV-0729B	J221	NA	3	
RC-SV-0707A	J220	NA	3	
RC-SV-0707B	J220	NA	3	

245

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
C. Primary Containment (Other Isolation Valves)				
1. Group 31 - Feedwater System				
(a) Feedwater Isolation Valves				M-41-1
Inside Check Valves				
AE-V003	P2A	NA	3	
AE-V007	P2B	NA	3	
Outside Check Valves (Air Assisted)				
HV-F074B (AE-V002)	P2A	NA	3	
HV-F074A (AE-V006)	P2B	NA	3	
2. Group 32 - Standby Liquid Control System				
Inside Check Valve				M-48-1
BH-V029	P18	NA	3	
3. Group 33 - Primary Containment Atmosphere Control System				
Containment Vacuum Breakers				M-57-1
Outside:				
GS-PSV-5032	P220	NA	3	
GS-PSV-5030	P219	NA	3	
4. Group 34 - Service Air System				M-15-0
Outside KA-V038	P27	NA	3	
Inside KA-V039	P27	NA	3	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
5. Group 35 - Breathing Air System				M-15-1
Inside KG-V016	P31	NA	3	
Outside KG-V034	P31	NA	3	
6. Group 36 - TIP Purge System				
Inside:				
Check Valve: SE-V006	P34F	NA	3	M-59-1
7. Group 37 - HPCI System				
Outside:				
HPCI Turbine Exhaust: FD-V004	P201	NA	4	M-55-1
8. Group 38 - RCIC System				
Outside:				
RCIC Turbine Exhaust: FC-V003	P207	NA	4	M-49-1
Vacuum Pump Discharge: FC-V010	P210	NA	4	M-49-1
9. Group 39 - RHR System				
(a) Thermal Relief Valves				M-51-1
Outside:				
Loop A: BC-PSV-F025A	P212B	NA	5	
Loop B: BC-PSV-F025B	P212A	NA	5	
Loop C: BC-PSV-F025C	P212B	NA	5	
Loop D: BC-PSV-F025D	P212A	NA	5	
(b) Jockey Pump Discharge Check Valves				M-51-1
Outside:				
Loops A & C: (BC-V206)	P212B	NA	4	
Loops B & D: (BC-V260)	P212A	NA	4	
(c) RHR Heat Exchanger Thermal Relief Valves				M-51-1
Outside:				
BC-PSV-4431A	P213B	NA	5	
BC-PSV-4431B	P213A	NA	5	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
(d) RHR Shutdown Cooling Suction Thermal Relief Valve				M-51-1
Inside:				
BC-PSV-4425	P3	NA	3	
(e) LPCI Injection Line Check Valves				M-51-1
Inside:				
HV-F041A (BC-V114)	P6C	NA	3	
HV-F041B (BC-V017)	P6B	NA	3	
HV-F041C (BC-V102)	P6D	NA	3	
HV-F041D (BC-V005)	P6A	NA	3	
(f) Shutdown Cooling Return Line Check Valves				M-51-1
Inside:				
HV-F050A (BC-V111)	P4B	NA	3	
HV-F050B (BC-V014)	P4A	NA	3	
(g) RHR Suppression Pool Return Valves				M-51-1
Outside:				
HV-F011A (BC-V126)	P212B	NA	4	
HV-F011B (BC-V026)	P212A	NA	4	
10. Group 40 - Core Spray System				
(a) Thermal Relief Valves				M-52-1
Outside:				
Loop A&C: BE-PSV-F012A	P217B	NA	5	
Loop B&D: BE-PSV-F012B	P217A	NA	5	
(b) Core Spray Injection Line Check Valves Inside:				M-52-1
HV-F006A (BE-V006)	P5B	NA	3	
HV-F006B (BE-V002)	P5A	NA	3	
11. Group 41 - Drywell Pressure Instrumentation				M-42-1
Outside:				
BB-V563	J6A	NA	6	
BB-V564	J8D	NA	6	
BB-V565	J7A	NA	6	
BB-V566	J10D	NA	6	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
12. Group 42 - Intergrated Leak Rate Testing System				M-60-1
Inside GP-V001	J36D	NA	3	
Outside GP-V002	J36D	NA	3	
Inside GP-V120	J36C	NA	3	
Outside GP-V122	J36C	NA	3	
Outside GP-V004	J209	NA	3	
Outside GP-V005	J209	NA	3	
13. Group 43 - Suppression Chamber Pressure Instrumentation				M-57-1
Outside				
GS-V044	J207	NA	6	
GS-V087	J208	NA	6	
14. Group 44 - Chilled Water System Thermal Relief Valves				M-87-1
Inside				
GB-PSV-9522A	P8B	NA	3	
GB-PSV-9522B	P38A	NA	3	
GB-PSV-9523A	P8A	NA	3	
GB-PSV-9523B	P38B	NA	3	
15. Group 45 - Recirculation Pump Seal Purge Line Check Valves				M-43-1
Inside				
BB-V043	P19	NA	3	
BB-V047	P20	NA	3	
D. Excess Flow Check Valves				
1. Group 46 - Nuclear Boiler				M-41-1
Outside				
BB-XV-3649	J5C	NA	6, 10	
AB-XV-3666A	J25A	NA	6	
AB-XV-3666B	J26A	NA	6	
AB-XV-3666C	J27A	NA	6	
AB-XV-3666D	J28A	NA	6	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
Outside				M-41-1
AB-XV-3667A	J22A	NA	6	
AB-XV-3667B	J22C	NA	6	
AB-XV-3667C	J21A	NA	6	
AB-XV-3667D	J21D	NA	6	
AB-XV-3668A	J22B	NA	6	
AB-XV-3668B	J22D	NA	6	
AB-XV-3668C	J21E	NA	6	
AB-XV-3668D	J21F	NA	6	
AB-XV-3669A	J25C	NA	6	
AB-XV-3669B	J26C	NA	6	
AB-XV-3669C	J27D	NA	6	
AB-XV-3669D	J28D	NA	6	
2. Group 47 - Nuclear Boiler Vessel Instrumentation				M-42-1
Outside				
BB-XV-3621	J3A	NA	6	
BB-XV-3725	J2C	NA	6	
BB-XV-3726A	J1350	NA	6	
BB-XV-3726B	J1353	NA	6	
BB-XV-3727A	J44	NA	6	
BB-XV-3727B	J41	NA	6	
BB-XV-3728A	J1351	NA	6	
BB-XV-3728B	J1354	NA	6	
BB-XV-3729A	J51	NA	6	
BB-XV-3729B	J42	NA	6	
BB-XV-3730A	J52	NA	6	
BB-XV-3730B	J43	NA	6	
BB-XV-3731A	J1352	NA	6	
BB-XV-3731B	J1355	NA	6	
BB-XV-3732A	J37A	NA	6	
BB-XV-3732B	J11A	NA	6	
BB-XV-3732C	J24E	NA	6	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
Outside				M-42-1
BB-XV-3732D	J11B	NA	6	
BB-XV-3732E	J37C	NA	6	
BB-XV-3732F	J40C	NA	6	
BB-XV-3732G	J37D	NA	6	
BB-XV-3732H	J40E	NA	6	
BB-XV-3732J	J37E	NA	6	
BB-XV-3732K	J11E	NA	6	
BB-XV-3732L	J14A	NA	6	
BB-XV-3732M	J40F	NA	6	
BB-XV-3732N	J14B	NA	6	
BB-XV-3732P	J12B	NA	6	
BB-XV-3732R	J14C	NA	6	
BB-XV-3732S	J12C	NA	6	
BB-XV-3732T	J14D	NA	6	
BB-XV-3732U	J40D	NA	6	
BB-XV-3732V	J14E	NA	6	
BB-XV-3732W	J12E	NA	6	
BB-XV-3734A	J50	NA	6	
BB-XV-3734B	J47	NA	6	
BB-XV-3734C	J14F	NA	6	
BB-XV-3734D	J12F	NA	6	
BB-XV-3737A	J38A	NA	6	
BB-XV-3737B	J16C	NA	6	
BB-XV-3738A	J13D	NA	6	
BB-XV-3738B	J38B	NA	6	
3. Group 48 - Reactor Recirculation System				M-43-1
Outside				
BB-XV-3783	J32B	NA	6	
BB-XV-3785	J32C	NA	6	
BB-XV-3787	J30C	NA	6	
BB-XV-3789	J30B	NA	6	
BB-XV-3801A	J18B	NA	6	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
Outside				M-43-1
BB-XV-3801B	J28B	NA	6	
BB-XV-3801C	J16E	NA	6	
BB-XV-3801D	J36E	NA	6	
BB-XV-3802A	J18F	NA	6	
BB-XV-3802B	J28F	NA	6	
BB-XV-3802C	J16F	NA	6	
BB-XV-3802D	J36F	NA	6	
BB-XV-3803A	J29F	NA	6	
BB-XV-3803B	J24A	NA	6	
BB-XV-3803C	J38C	NA	6	
BB-XV-3803D	J34D	NA	6	
BB-XV-3804A	J29D	NA	6	
BB-XV-3804B	J24B	NA	6	
BB-XV-3804C	J38F	NA	6	
BB-XV-3804D	J34E	NA	6	
4. Group 49 - Reactor Recirculation System - Cont'd.				M-43-1
Outside				
BB-XV-3820	J32E	NA	6	
BB-XV-3821	J32F	NA	6	
BB-XV-3826	J34B	NA	6	
BB-XV-3827	J23C	NA	6	
5. Group 50 - Reactor Water Cleanup				M-44-1
Outside				
BG-XV-3882	J24C	NA	6	
BG-XV-3884A	J19D	NA	6	
BG-XV-3884B	J34A	NA	6	
BG-XV-3884C	J19E	NA	6	
BG-XV-3884D	J34C	NA	6	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
6. Group 51 - Reactor Core Isolation Cooling System				M-49-1
Outside				
FC-XV-4150A	J20A	NA	6	
FC-XV-4150B	J40B	NA	6	
FC-XV-4150C	J20B	NA	6	
FC-XV-4150D	J40A	NA	6	
7. Group 52 - Residual Heat Removal System				M-51-1
Outside				
BC-XV-4411A	J33A	NA	6	
BC-XV-4411B	J23B	NA	6	
BC-XV-4411C	J35A	NA	6	
BC-XV-4411D	J36B	NA	6	
BC-XV-4429A	J33D	NA	6	
BC-XV-4429B	J23A	NA	6	
BC-XV-4429C	J35C	NA	6	
BC-XV-4429D	J36A	NA	6	
8. Group 53 - Core Spray System				M-52-1
Outside				
BE-XV-F018A	J19C	NA	6	
BE-XV-F018B	J30F	NA	6	
9. Group 54 - High Pressure Coolant Injection System				M-55-1
Outside				
FD-XV-4800A	J19A	NA	6	
FD-XV-4800B	J29A	NA	6	
FD-XV-4800C	J19B	NA	6	
FD-XV-4800D	J29B	NA	6	

TABLE 3.6.3-1

PRIMARY CONTAINMENT ISOLATION VALVES

NOTES

NOTATION

1. Main Steam Isolation Valves are sealed with a seal system that maintains a positive pressure of 5 psig above reactor pressure. Leakage is in-leakage and is not added to 0.60 La allowable leakage.*
2. Containment Isolation Valves are sealed with a water seal from the HPCI and/or RCIC system to form the long-term seal boundary of the feedwater lines. The valves are tested with water at 1.10 Pa, 52.9 psig, to ensure the seal boundary will prevent by-pass leakage. Seal boundary liquid leakage will be limited to 10 gpm.
3. Containment Isolation Valve, Type C gas test at Pa, 48.1 psig. Leakage added to 0.60La allowable leakage.
4. Containment Isolation Valve, Type C water test at Pa, 48.1 psig Δ P. Leakage added to 10 gpm allowable leakage.
5. Containment boundary is discharge nozzle of relief valve, leakage tested during Type A test.*
6. Drywell and suppression chamber pressure and level instrument root valves and excess flow check valves, leakage tested during Type A.*
7. Explosive shear valves (SE-V021 through SE-V025) not Type C tested.*
8. Surveillances to be performed per Specification 3.6.1.8.
9. All valve I.D. numbers are preceded by a numeral 1 which represents an Unit 1 valve.
10. The reactor vessel head seal leak detection line (penetration J5C) excess flow check valve (BB-XV-3649) is not subject to OPERABILITY testing. This valve will not be exposed to primary system pressure except under the unlikely conditions of a seal failure where it could be partially pressurized to reactor pressure. Any leakage path is restricted at the source; therefore, this valve need not be OPERABILITY tested.

---

*Exemption to Appendix J of 10 CFR Part 50.

3/4.6.4 VACUUM RELIEF

SUPPRESSION CHAMBER - DRYWELL VACUUM BREAKERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 All suppression chamber - drywell vacuum breakers shall be OPERABLE and closed.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one of the above vacuum breakers inoperable for opening but known to be closed, restore the inoperable vacuum breaker to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one or more suppression chamber - drywell vacuum breaker(s) open, close the open vacuum breaker(s) within 2 hours; or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one of the position indicators of any suppression chamber - drywell vacuum breaker inoperable:

    1. Verify the other position indicator in the pair to be OPERABLE within 2 hours and at least once per 14 days thereafter, or

    2. Verify the vacuum breaker(s) with the inoperable position indicator to be closed by conducting a test which demonstrates that the ΔP is maintained at greater than or equal to 0.5 psi for one hour without makeup within 24 hours and at least once per 14 days thereafter.

    Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each suppression chamber - drywell vacuum breaker shall be:

a. Verified closed at least once per 7 days.

b. Demonstrated OPERABLE:

1. At least once per 31 days and within 2 hours after any discharge of steam to the suppression chamber from the safety-relief valves, by cycling each vacuum breaker through at least one complete cycle of full travel.

2. At least once per 31 days by verifying both position indicators OPERABLE by observing expected valve movement during the cycling test.

3. At least once per 18 months by;

a) Verifying the opening setpoint, from the closed position, to be less than or equal to 0.20 psid, and

b) Verifying both position indicators OPERABLE by performance of a CHANNEL CALIBRATION.

256

REACTOR BUILDING - SUPPRESSION CHAMBER VACUUM BREAKERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 Both reactor building - suppression chamber vacuum breaker assemblies consisting of a vacuum breaker valve and a butterfly isolation valve shall be OPERABLE and closed.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one valve of a reactor building - suppression chamber vacuum breaker assembly inoperable for opening but known to be closed, restore the inoperable vacuum breaker assembly valve to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one valve of a reactor building - suppression chamber vacuum breaker assembly open, verify the other vacuum breaker assembly valve in the line to be closed within 2 hours; restore the open vacuum breaker assembly valve to the closed position within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With the position indicator of any reactor building - suppression chamber vacuum breaker assembly valve inoperable, restore the inoperable position indicator to OPERABLE status within 14 days or verify the affected vacuum breaker assembly valve to be closed at least once per 24 hours by a visual inspection. Otherwise, declare the vacuum breaker assembly valve inoperable or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.2 Both reactor building - suppression chamber vacuum breaker assemblies shall be:

a. Verified closed at least once per 7 days.

b. Demonstrated OPERABLE:

1. At least once per 31 days by:

    a) Cycling each vacuum breaker assembly valve through at least one complete cycle of full travel.

    b) Verifying the position indicators on each assembly valve OPERABLE by observing expected valve movement during the cycling test.

2. At least once per 18 months by:

    a) Demonstrating that the force required to open each vacuum breaker valve does not exceed the equivalent of 0.25 psid.

    b) Visual inspection.

257

SURVEILLANCE REQUIREMENTS (Continued)

c) Verifying the position indicators on each assembly valve OPERABLE by performance of a CHANNEL CALIBRATION.

d) Verifying the instrument actuation system for the inboard isolation valve auto open control system OPERABLE by performance of a CHANNEL CALIBRATION.

3/4.6.5 SECONDARY CONTAINMENT

SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and *.

ACTION:

Without SECONDARY CONTAINMENT INTEGRITY:

a. In OPERATIONAL CONDITION 1, 2 or 3, restore SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In Operational Condition *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying at least once per 24 hours that the reactor building is at a negative pressure.

b. Verifying at least once per 31 days that:

  1. All secondary containment equipment hatches and blowout panels are closed and sealed.

  2. a. For double door arrangements, at least one door in each access to the secondary containment is closed.

     b. For single door arrangements, the door in each access to the secondary containment is closed except for routine entry and exit.

  3. All secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers/valves secured in position.

c. At least once per 18 months:

  1. Verifying that four filtration recirculation and ventilation system (FRVS) recirculation units and one ventilation unit of the filtration recirculation and ventilation system will draw down the secondary containment to greater than or equal to 0.25 inches of vacuum water gauge in less than or equal to 375 seconds, and

---

*When irradiated fuel is being handled in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

SURVEILLANCE REQUIREMENTS (Continued)

2. Operating four filtration recirculation and ventilation system (FRVS) recirculation units and one ventilation unit of the filtration recirculation and ventilation system for four hours and maintaining greater than or equal to 0.25 inches of vacuum water gauge in the secondary containment at a flow rate not exceeding 3324 CFM.

SECONDARY CONTAINMENT AUTOMATIC ISOLATION DAMPERS

LIMITING CONDITION FOR OPERATION

3.6.5.2 The secondary containment ventilation system (RBVS) automatic isolation dampers shown in Table 3.6.5.2-1 shall be OPERABLE with isolation times less than or equal to the times shown in Table 3.6.5.2-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and *.

ACTION:

With one or more of the secondary containment ventilation system automatic isolation dampers shown in Table 3.6.5.2-1 inoperable, maintain at least one isolation damper OPERABLE in each affected penetration that is open and within 8 hours either:

a. Restore the inoperable dampers to OPERABLE status, or

b. Isolate each affected penetration by use of at least one deactivated damper secured in the isolation position, or

c. Isolate each affected penetration by use of at least one closed manual valve or blind flange.

Otherwise, in OPERATIONAL CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Otherwise, in Operational Condition *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.2 Each secondary containment ventilation system automatic isolation damper shown in Table 3.6.5.2-1 shall be demonstrated OPERABLE:

a. Prior to returning the damper to service after maintenance, repair or replacement work is performed on the damper or its associated actuator, control or power circuit by cycling the damper through at least one complete cycle of full travel and verifying the specified isolation time.

b. During COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a containment isolation test signal each isolation damper actuates to its isolation position.

c. By verifying the isolation time to be within its limit at least once per 92 days.

---

*When irradiated fuel is being handled in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

261

TABLE 3.6.5.2-1

SECONDARY CONTAINMENT VENTILATION SYSTEM AUTOMATIC ISOLATION DAMPERS

ISOLATION GROUP NO. 19

DAMPER FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
1. Reactor Building Ventilation Supply Damper HD-9370A	10
2. Reactor Building Ventilation Supply Damper HD-9370B	10
3. Reactor Building Ventilation Exhaust Damper HD-9414A	10
4. Reactor Building Ventilation Exhaust Damper HD-9414B	10

FILTRATION, RECIRCULATION AND VENTILATION SYSTEM (FRVS)

LIMITING CONDITION FOR OPERATION

3.6.5.3 Five FRVS recirculation units and two FRVS ventilation units shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and *.

ACTION:

a. With one of the above required FRVS recirculation units or one of the above required FRVS ventilation units inoperable, restore the inoperable unit to OPERABLE status within 7 days, or:

   1. In OPERATIONAL CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   2. In Operational Condition *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

b. With three FRVS recirculation units or both ventilation units inoperable in Operational Condition *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS or operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3. are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.3 Each of the six FRVS recirculation and two ventilation units shall be demonstrated OPERABLE:

a. At least once per 14 days by verifying that the water seal bucket traps have a water seal and making up any evaporative losses by filling the traps to the overflow.

b. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters on in order to reduce the buildup of moisture on the carbon adsorbers and HEPA filters.

---

*When irradiated fuel is being handled in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the subsystem by:

   1. Verifying that the subsystem satisfies the in-place penetration testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rates are 30,000 cfm ± 10% for each FRVS recirculation unit, and 9,000 cfm ± 10% for each FRVS ventilation unit.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803; and

   3. Verifying a subsystem flow rate of 30,000 cfm ± 10% for each FRVS recirculation unit and 9,000 cfm ±10% for each FRVS ventilation unit during system operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 1.0% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803.

e. At least once per 18 months by:

   1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 8 inches Water Gauge in the recirculation filter train and less than 5 inches Water Gauge in the ventilation unit while operating the filter train at a flow rate of 30,000 cfm ± 10% for each FRVS recirculation unit and 9,000 cfm ± 10% for each FRVS ventilation unit.

   2. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

      a. Manual initiation from the control room, and

SURVEILLANCE REQUIREMENTS (Continued)

b Simulated automatic initiation signal.

3. Verifying that the heaters dissipate 100 ± 10 kw for each recirculation unit and 32 ± 3 kw for each ventilation unit when tested in accordance with ANSI N510-1980, and verifying humidity is maintained less than or equal to 70% relative humidity through the carbon adsorbers by performance of a channel calibration of the humidity control instrumentation.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the inplace penetration testing acceptance criteria of less than 0.05% in accordance with Regulatory Position C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2 March 1978, while operating the system at a flow rate of 30,000 cfm ± 10% for each FRVS recirculation unit and 9,000 cfm ± 10% for each FRVS ventilation unit.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the inplace penetration testing acceptance criteria of less than 0.05% in accordance with Regulatory Position C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 30,000 cfm ± 10% for each FRVS recirculation unit and 9,000 cfm ± 10% for each FRVS ventilation unit.

DRYWELL AND SUPPRESSION CHAMBER OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.6.6.2 The drywell and suppression chamber atmosphere oxygen concentration shall be less than 4% by volume.

APPLICABILITY: OPERATIONAL CONDITION 1*, during the time period:

a. Within 24 hours after THERMAL POWER is greater than 15% of RATED THERMAL POWER, following startup, to

b. Within 24 hours prior to reducing THERMAL POWER to less than 15% of RATED THERMAL POWER, preliminary to a scheduled reactor shutdown.

ACTION:

With the drywell and/or suppression chamber oxygen concentration exceeding the limit, restore the oxygen concentration to within the limit within 24 hours or be in at least STARTUP within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.2 The drywell and suppression chamber oxygen concentration shall be verified to be within the limit within 24 hours after THERMAL POWER is greater than 15% of RATED THERMAL POWER and at least once per 7 days thereafter.

---

*See Special Test Exception 3.10.5.

3/4.7.1 SERVICE WATER SYSTEMS

SAFETY AUXILIARIES COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.1 At least the following independent safety auxiliaries cooling system (SACS) subsystems, with each subsystem comprised of:

a. Two OPERABLE SACS pumps, and

b. An OPERABLE flow path consisting of a closed loop through the SACS heat exchangers and SACS pumps and to associated safety related equipment

shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2 and 3, two subsystems.

b. In OPERATIONAL CONDITION 4, 5, and ** the subsystems associated with systems and components required OPERABLE by Specification 3.4.9.1, 3.4.9.2, 3.5.2, 3.8.1.2, 3.9.11.1 and 3.9.11.2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5, and **.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3:

1. With one SACS pump or heat exchanger inoperable, restore the inoperable pump or heat exchanger to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one SACS subsystem otherwise inoperable, realign the affected diesel generators to the OPERABLE SACS subsystem within 2 hours, and restore the inoperable subsystem to OPERABLE status with at least one OPERABLE pump and heat exchanger within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one SACS pump or heat exchanger in each subsystem inoperable, immediately initiate measures to place the unit in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. With both SACS subsystems otherwise inoperable, immediately initiate measures to place the unit in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN* in the following 24 hours.

---

*Whenever both SACS subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

**When handling irradiated fuel in the secondary containment.

268

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b. In OPERATIONAL CONDITION 3 or 4 with the SACS subsystem, which is associated with an RHR loop required OPERABLE by Specification 3.4.9.1 or 3.4.9.2, inoperable, declare the associated RHR loop inoperable and take the ACTION required by Specification 3.4.9.1 or 3.4.9.2, as applicable.

c. In OPERATIONAL CONDITION 4 or 5 with the SACS subsystem, which is associated with safety related equipment required OPERABLE by Specification 3.5.2, inoperable, declare the associated safety related equipment inoperable and take the ACTION required by Specification 3.5.2.

d. In OPERATIONAL CONDITION 5 with the SACS subsystem, which is associated with an RHR loop required OPERABLE by Specification 3.9.11.1 or 3.9.11.2, inoperable, declare the associated RHR system inoperable and take the ACTION required by Specification 3.9.11.1 or 3.9.11.2, as applicable.

e. In OPERATIONAL CONDITION 4, 5, or **, with one SACS subsystem, which is associated with safety related equipment required OPERABLE by Specification 3.8.1.2, inoperable, realign the associated diesel generators within 2 hours to the OPERABLE SACS subsystem, or declare the associated diesel generators inoperable and take the ACTION required by Specification 3.8.1.2. The provisions of Specification 3.0.3 are not applicable.

f. In OPERATIONAL CONDITION 4, 5, or **, with only one SACS pump and heat exchanger and its associated flowpath OPERABLE, restore at least two pumps and two heat exchangers and associated flowpaths to OPERABLE status within 72 hours or, declare the associated safety related equipment inoperable and take the associated ACTION requirements.

SURVEILLANCE REQUIREMENTS

4.7.1.1 At least the above required safety auxiliaries cooling system subsystems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown by verifying that: 1) Each automatic valve servicing safety-related equipment actuates to its correct position on the appropriate test signal(s), and 2) Each pump starts automatically when its associated diesel generator automatically starts.

STATION SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least the following independent station service water system loops, with each loop comprised of:

a. Two OPERABLE station service water pumps, and

b. An OPERABLE flow path capable of taking suction from the Delaware River (ultimate heat sink) and transferring the water to the SACS heat exchangers,

shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2 and 3, two loops.

b. In OPERATIONAL CONDITION 4, 5 and *, one loop.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5 and *.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3:

1. With one station service water pump inoperable, restore the inoperable pump to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
2. With one station service water pump in each loop inoperable, restore at least one inoperable pump to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
3. With one station service water system loop otherwise inoperable, restore the inoperable station service water system loop to OPERABLE status with at least one OPERABLE pump within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4 or 5:

With only one station service water pump and its associated flowpath OPERABLE, restore at least two pumps with at least one flow path to OPERABLE status within 72 hours or declare the associated SACS subsystem inoperable and take the ACTION required by Specification 3.7.1.1.

c. In OPERATIONAL CONDITION *:

With only one station service water pump and its associated flowpath OPERABLE, restore at least two pumps with at least one flow path to OPERABLE status within 72 hours or declare the associated SACS subsystem inoperable and take the ACTION required by Specification 3.7.1.1. The provisions of Specification 3.0.3 are not applicable.

---

*When handling irradiated fuel in the secondary containment.

270

SURVEILLANCE REQUIREMENTS

4.7.1.2 At least the above required station service water system loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic), servicing safety related equipment that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown, by verifying that:

   1. Each automatic valve servicing non-safety related equipment actuates to its isolation position on an isolation test signal.

   2. Each pump starts automatically when its associated diesel generator automatically starts.

ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The ultimate heat sink (Delaware River) shall be OPERABLE with:

a. A minimum river water level at or above elevation -13'0 Mean Sea Level, USGS datum (76'0 PSE&G datum), and

b. An average river water temperature of less than or equal to 90.5°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5 and *.

ACTION:

With the requirements of the above specification not satisfied:

a. In OPERATIONAL CONDITIONS 1, 2 or 3, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. In OPERATIONAL CONDITIONS 4 or 5, declare the SACS system and the station service water system inoperable and take the ACTION required by Specification 3.7.1.1 and 3.7.1.2.

c. In Operational Condition *, declare the plant service water system inoperable and take the ACTION required by Specification 3.7.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The ultimate heat sink shall be determined OPERABLE:

a. By verifying the river water level to be greater than or equal to the minimum limit at least once per 24 hours.

b. By verifying river water temperature to be within its limit:

1) at least once per 24 hours when the river water temperature is less than or equal to 85°F.

2) at least once per 6 hours when the river water temperature is greater than 85°F.

---

*When handling irradiated fuel in the secondary containment.

BEST COPY AVAILABLE

3/4.7.2 CONTROL ROOM EMERGENCY FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 Two independent control room emergency filtration system subsystems shall be OPERABLE with each subsystem consisting of:

a) One control room supply unit,
b) One filter train, and
c) One control room return air fan.

APPLICABILITY: All OPERATIONAL CONDITIONS and *.

ACTION:

a. In OPERATIONAL CONDITION 1, 2 or 3 with one control room emergency filtration subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4, 5 or *:

  1. With one control room emergency filtration subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or initiate and maintain operation of the OPERABLE subsystem in the pressurization/recirculation mode of operation.

  2. With both control room emergency filtration subsystems inoperable, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

c. The provisions of Specification 3.0.3 are not applicable in Operational Condition *.

SURVEILLANCE REQUIREMENTS

4.7.2 Each control room emergency filtration subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 85°F[#].

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, the control area chilled water pump, flow

---

*When irradiated fuel is being handled in the secondary containment.

[#] This does not require starting the non-running control emergency filtration subsystem.

through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters on in order to reduce the buildup of moisture on the carbon adsorbers and HEPA filters.

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the subsystem by:

  1. Verifying that the subsystem satisfies the in-place penetration testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system filter train flow rate is 4000 cfm ± 10%.

  2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 0.175% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ASTM D3803 with a 4 inch bed; and

  3. Verifying a subsystem filter train flow rate of 4000 cfm ± 10% during subsystem operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, by showing a methyl iodide penetration of less than 0.175% when tested at a temperature of 30°C and at a relative humidity of 70% in accordance with ATSM D3803 with a 4 inch bed.

e. At least once per 18 months by:

  1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 7.5 inches Water Gauge while operating the filter train subsystem at a flow rate of 4000 cfm ± 10%.

  2. Verifying with the control room hand switch in the recirculation mode that on each of the below recirculation mode actuation test signals, the subsystem automatically switches to the isolation mode of operation and the isolation dampers close within 5 seconds:

SURVEILLANCE REQUIREMENTS (Continued)

a) High Drywell Pressure
b) Reactor Vessel Water Level Low Low Low, Level 1
c) Control room ventilation radiation monitors high.

3. Verifying with the control room hand switch in the outside air mode that on each of the below pressurization mode actuation test signals, the subsystem automatically switches to the pressurization mode of operation and the control room is maintained at a positive pressure of at least 1/8 inch water gauge relative to adjacent areas during subsystem operation at a flow rate less than or equal to 1000 cfm:

   a) High Drywell Pressure

   b) Reactor Vessel Water Level Low Low Low, Level 1

   c) Control room ventilation radiation monitors high.

4. Verifying that the heaters dissipate 13 ± 1.3 Kw when tested in accordance with ANSI N510-1980 and verifying humidity is maintained less than or equal to 70% humidity through the carbon adsorbers by performance of a channel calibration of the humidity control instrumentation.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the inplace penetration testing acceptance criteria of less than 0.05% in accordance with Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, March 1978, while operating the system at a flow rate of 4000 cfm ± 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the inplace penetration testing acceptance criteria of less than 0.05% in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 4000 cfm ± 10%.

3/4.7.3 FLOOD PROTECTION

LIMITING CONDITION FOR OPERATION

3.7.3 Flood protection shall be provided for all safety related systems, components and structures when the water level of the Delaware River reaches 6.0 feet Mean Sea Level (MSL) USGS datum (95.0 feet PSE&G datum) at the Service Water Intake Structure.

APPLICABILITY: At all times.

ACTION:

a. With severe storm warnings from the National Weather Service which may impact Artificial Island in effect or with the water level at the service water intake structure above elevation 6.0 feet MSL USGS datum (95.0 feet PSE&G datum), initiate and complete:

   1. The closing of all service water intake structure watertight perimeter flood doors identified in Table 3.7.3-1 within 1 hour, and

   2. The closing of all power block watertight perimeter flood doors identified in Table 3.7.3-1 within 1.5 hours.

   Once closed, all access through the doors shall be administratively controlled.

b. With the water level at the service water intake structure above elevation 10.5 feet MSL USGS datum (99.5 feet PSE&G datum), be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 The water level at the service water intake structure shall be determined to be within the limit by:

a. Measurement at least once per 24 hours when the water level is below elevation 6.0 MSL USGS datum (95.0 feet PSE&G datum), and

b. Measurement at least once per 4 hours when severe storm warnings from the National Weather Service which may impact Artificial Island are in effect.

c. Measurement at least once per hour when the water level is equal to or above elevation 6.0 MSL USGS datum (95.0 feet PSE&G datum).

TABLE 3.7.3-1

PERIMETER FLOOD DOORS

INTAKE STRUCTURE DOORS

Water tight door 1
Water tight door 2
Water tight door 3
Water tight door 4
Water tight door 5
Water tight door 6
Water tight door 7
Water tight door 8

POWER BLOCK DOORS and HATCH

Doors & Hatch		Location
Hatch	Exterior	45; K
S-13	"	45.5; L
3340B	"	44; M
3337B	"	44; Md
6312	"	45.4; T
6323B	"	45.4; U
5315A	"	29.9; X
5315C	"	29; X
4323A	"	13.6; U
4304	"	13.6; U
3301A	"	13.6; Md
3305B	"	13.6; L
3315B	Interior-102'	25; H
3329A	"	27; H
3331B	"	35; H
3209A	Interior	26; H

3/4.7.4 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of automatically taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:
With the RCIC system inoperable, operation may continue provided the HPCI system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 150 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

   2. Verifying that each valve, manual, power operated or automatic in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

   3. Verifying that the pump flow controller is in the correct position.

b. When tested pursuant to Specification 4.0.5 by verifying that the RCIC pump develops a flow of greater than or equal to 600 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is being supplied to the turbine at 1000 + 20, - 80 psig.*

---

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation and restart[#] and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded.

2. Verifying that the system will develop a flow of greater than or equal to 600 gpm in the test flow path when steam is supplied to the turbine at a pressure of 150 + 15, - 0 psig.*

3. Verifying that the suction for the RCIC system is automatically transferred from the condensate storage tank to the suppression pool on a condensate storage tank water level-low signal.

---

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the tests.

[#] Automatic restart on a low water level signal which is subsequent to a high water level trip.

3/4.7.5 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.5 All snubbers shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1,2, and 3. OPERATIONAL CONDITIONS 4 and 5 for snubbers located on systems required OPERABLE in those OPERATIONAL CONDITIONS.

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.5.g on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.5 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these groups (inaccessible and accessible) may be inspected independently according to the schedule below. The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing power operation and shall include all snubbers. If all snubbers of each type are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection shall be performed at the first refueling outage. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

BEST COPY AVAILABLE

280

No. Inoperable Snubbers of Each Type per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection period, providing that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type on that system that may be generically susceptible; or (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specifications 4.7.4.f. For those snubbers common to more than one system, the OPERABILITY of such snubbers shall be considered in assessing the surveillance schedule.

d. Transient Event Inspection

An inspection shall be performed of all snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients, as determined from a review of operational data or a visual inspection of the systems, within 72 hours for accessible systems and 6 months for inaccessible systems following this determination. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement, or (2) evaluation of in-place snubber piston setting.

---

*The inspection interval for each type of snubber shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found.

#The provisions of Specification 4.0.2 are not applicable.

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of snubbers shall be tested using one of the following sample plans for each type of snubber. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.5.f., an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time, providing all snubbers tested with the failed equipment during the day of equipment failure are retested; or

2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7.5-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.5.f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of testing "N" snubbers, the results shall be plotted on Figure 4.7.5-1. If at any time the point plotted falls on or above the "Reject" line all snubbers of that type shall be functionally tested. If at any time the point plotted falls on or below the "Accept" line, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time, providing all snubbers tested with the failed equipment during the day of equipment failure are retested; or

3) An initial representative sample of 55 snubbers of each type shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, $1 + C/2$, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation $N = 55(1 + C/2)$. Each snubber point should be plotted when "N" snubbers have been tested. If the

282

point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls on or below the "Accept" line or all the snubbers of that type have been tested. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time, providing all snubbers tested with the failed equipment during the day of equipment failure are retested.

The representative sample selected for the initial function test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure as far as practical that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same locations as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan, and failure of this functional test shall not be the sole cause for increasing the sample size under the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional testing results shall be reviewed at the time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression;

2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range (hydraulic snubbers only);

3) For mechanical snubbers, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

4) For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the

283

OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.5.e. for snubbers not meeting the functional test acceptance criteria.

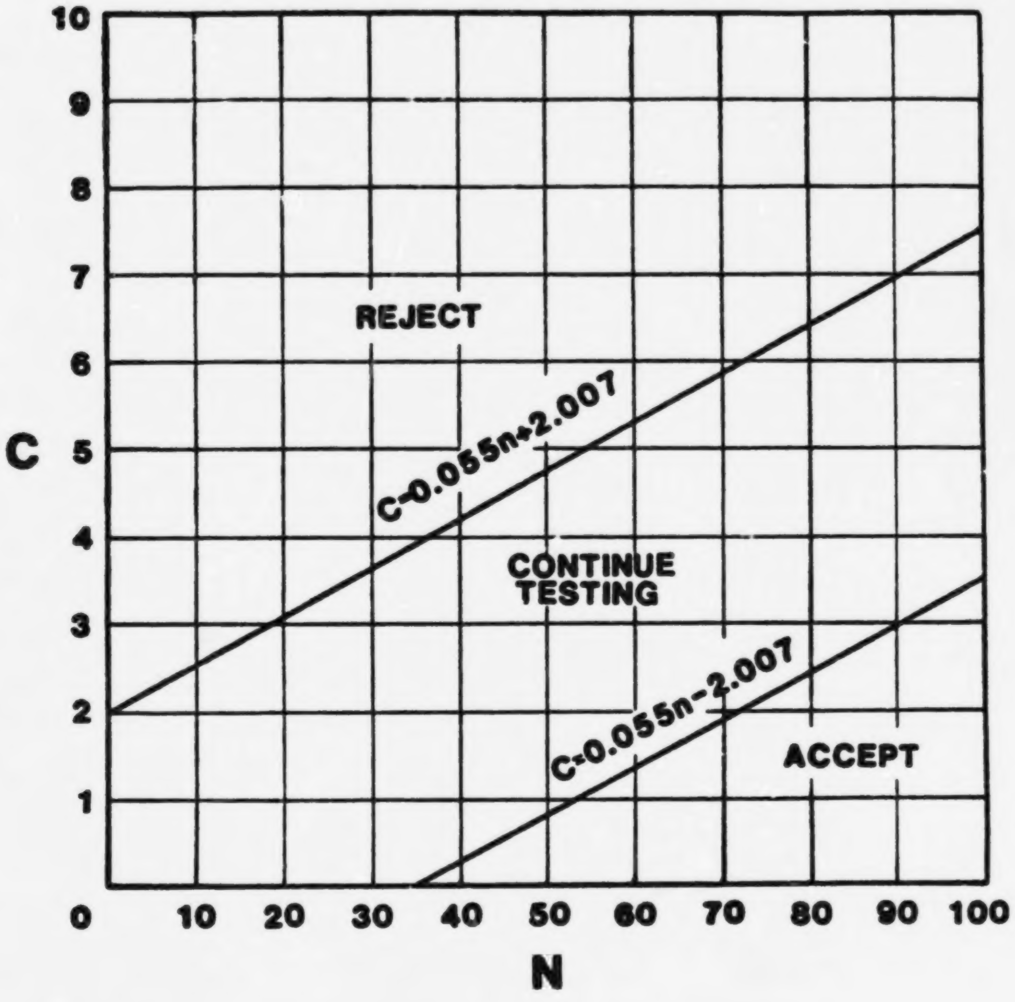
h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test result shall be tested to meet the functional test criteria before installation in the unit. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent service, and the freedom-of-motion test must have been performed within 12 months before being installed in the unit.

i. Snubber Service Life Replacement Program

The service life of all snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.3.

284



SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST

Figure 4.7.5-1

285

3/4.7.5 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.6 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limit, withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.6.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.6.2 Test Frequencies - Each category of sealed sources, excluding startup sources and fission detectors previously subjected to core flux, shall be tested at the frequency described below.

a. Sources in use - At least once per six months for all sealed sources containing radioactive material:

   1. With a half-life greater than 30 days, excluding Hydrogen 3, and

   2. In any form other than gas.

286

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.6.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

287

3/4.7.7 MAIN TURBINE BYPASS SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 The main turbine bypass system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION: With the main turbine bypass system inoperable, restore the system to OPERABLE status within 2 hours or reduce THERMAL POWER to less than or equal to 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.7.7 The main turbine bypass system shall be demonstrated OPERABLE at least once per:

a. 31 days by cycling each turbine bypass valve through at least one complete cycle of full travel, and

b. 18 months by:

   1. Performing a system functional test which includes simulated automatic actuation and verifying that each automatic valve actuates to its correct position.

   2. Demonstrating TURBINE BYPASS SYSTEM RESPONSE TIME meets the following requirements when measured from the initial movement of the main turbine stop or control valve:

      a) 80% of turbine bypass system capacity shall be established in less than or equal to 0.3 second.

      b) Bypass valve opening shall start in less than or equal to 0.1 second.

288

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

A.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Four separate and independent diesel generators, each with:

   1. A separate fuel oil day tank containing a minimum of 200 gallons of fuel,
   2. A separate fuel storage system consisting of two storage tanks containing a minimum of 48,800 gallons of fuel, and
   3. A separate fuel transfer pump for each storage tank.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one offsite circuit of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If any diesel generator has not been successfully tested within the past 24 hours, demonstrate its OPERABILITY by performing Surveillance Requirement 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 for each such diesel generator separately within 24 hours. Restore the inoperable offsite circuit to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the above required A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the diesel generator became inoperable due to any cause other than preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 separately for each diesel generator within 24 hours*; restore the inoperable diesel generator to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one offsite circuit of the above required A.C. sources and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and

---

*This test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY.

289

ACTION: (Continued)

at least once per 8 hours thereafter. If a diesel generator became inoperable due to any causes other than preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generators separately for each diesel generator by performing Surveillance Requirement 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 within 24 hours.* Restore at least two offsite circuits and all four of the above required diesel generators to OPERABLE status within 72 hours from time of the initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. A successful test(s) of diesel generator OPERABILITY per Surveillance Requirement 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 performed under this ACTION statement for the OPERABLE diesel generators satisfies the diesel generator test requirements of ACTION Statement b.

d. With both of the above required offsite circuits inoperable, demonstrate the OPERABILITY of all of the above required diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 separately for each diesel generator within 8 hours unless the diesel generators are already operating; restore at least one of the above required offsite circuits to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours. With only one offsite circuit restored to OPERABLE status, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. A successful test(s) of diesel generator OPERABILITY per Surveillance Requirement 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 performed under this ACTION statement for the OPERABLE diesel generators satisfies the diesel generator test requirements of ACTION statement a.

e. With two diesel generators of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the above required A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter and demonstrate the OPERABILITY of the remaining diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 separately for each diesel generator within 8 hours.* Restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Restore both of the inoperable diesel generators to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in

---

*This test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

COLD SHUTDOWN within the following 24 hours. A successful test(s) of diesel generator OPERABILITY per Surveillance Requirement 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 performed under this ACTION statement for the OPERABLE diesel generators satisfies the diesel generator test requirements of ACTION Statements a and b.

f. With two diesel generators of the above required A.C. electrical power sources inoperable, in addition to ACTION e., above, verify within 2 hours that all required systems, subsystems, trains, components, and devices that depend on the remaining diesel generators as a source of emergency power are also OPERABLE; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

g. With one offsite circuit and two diesel generators of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter and demonstrate the OPERABILITY of the remaining diesel generators by performing Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 separately for each diesel generator within 8 hours.* Restore at least one of the above required inoperable A.C. sources to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Restore the inoperable offsite circuit and both of the inoperable diesel generators to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

---

*This test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY.

291

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments and indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring, manually and automatically, unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each of the above required diesel generators shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8.1.1.2-1 on a STAGGERED TEST BASIS by:

  1. Verifying the fuel level in the fuel oil day tank.

  2. Verifying the fuel level in the fuel oil storage tank.

  3. Verifying the fuel transfer pump starts and transfers fuel from the storage system to the fuel oil day tank.

  4. Verifying the diesel starts from ambient conditions and accelerates to at least 514 rpm in less than or equal to 10 seconds after receipt of the start signal.* The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after receipt of the start signal. The diesel generator shall be started for this test by using one of the following signals:

    a) Manual.

    b) Simulated loss of offsite power by itself.

    c) Simulated loss of offsite power in conjunction with an ESF actuation test signal.

    d) An ESF actuation test signal by itself.

  5. Verifying the diesel generator is synchronized, loaded to between 4300 and 4400** kw in less than or equal to 130 seconds,* and operates with this load for at least 60 minutes.

---

*The diesel generator start (10 sec) and subsequent loading (130 sec) from ambient conditions shall be performed at least once per 184 days in these surveillance tests. All other engine starts and loading for the purpose of this surveillance testing may be preceded by an engine prelube period and/or other warmup procedures recommended by the manufacturer so that mechanical stress and wear on the diesel engine is minimized.

**This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band shall not invalidate the test; the loads, however, shall not be less than 4300 kw nor greater than 4430 kw.

SURVEILLANCE REQUIREMENTS (Continued)

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

7. Verifying the pressure in all diesel generator air start receivers to be greater than or equal to 380 psig.

8. Verifying the lube oil pressure, temperature and differential pressure across the lube oil filters to be within manufacturer's specifications.

b. At least once per 31 days by visually examining a sample of lube oil from the diesel engine to verify absence of water.

c. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the fuel oil day tank.

d. At least once per 92 days by removing accumulated water from the fuel oil storage tanks.

e. At least once per 31 days by performing a functional test on the emergency load sequencer to verify operability.

f. At least once per 92 days and from new fuel oil prior to addition to the storage tanks by obtaining a sample in accordance with ASTM-D270-1975 and by verifying that the sample meets the following minimum requirements and is tested within the specified time limits:

1. As soon as sample is taken or from new fuel prior to addition to the storage tank, as applicable, verify in accordance with the tests specified in ASTM-D975-77 that the sample has:

    a) A water and sediment content of less than or equal to 0.05 volume percent.

    b) A kinematic viscosity @ 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes or a Saybolt Second Universal (SSU) viscosity at 100°F of greater than or equal to 32 SSU but less than or equal to 40.1 SSU.

    c) A specific gravity as specified by the manufacturer as API gravity @ 60°F of greater than or equal to 28 degrees but less than or equal to 42 degrees.

2. Within one week after obtaining the sample, verify an impurity level of less than 2 mg of insolubles per 100 ml. when tested in accordance with ASTM-D2274-70.

SURVEILLANCE REQUIREMENTS (Continued)

    3. Within 2 weeks after obtaining the sample, verify that the other properties specified in Table 1 of ASTM-D975-77 and Regulatory Guide 1.137, Position 2.a, are met when tested in accordance with ASTM-D975-77.

g. At least once per 2 months, by verifying the buried fuel oil transfer piping's cathodic protection system is OPERABLE and at least once per year by subjecting the cathodic protection system to a performance test.

h. At least once per 18 months#, during shutdown, by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

2. Verifying the diesel generator capability to reject a load of greater than or equal to that of the RHR pump motor (1003 kW) for each diesel generator while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 1.2 Hz.

3. Verifying the diesel generator capability to reject a load of 4430 kW without tripping. The generator voltage shall not exceed 4580 volts during and following the load rejection.

4. Simulating a loss of offsite power by itself, and:

    a) Verifying loss of power is detected and deenergization of the emergency busses and load shedding from the emergency busses.

    b) Verifying the diesel generator starts* on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds after receipt of the start signal, energizes the autoconnected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

---

*This diesel generator start (10 sec) and subsequent loading (130 sec) from ambient conditions may be preceded by an engine prelube period and/or other warmup procedures recommended by the manufacturer so that mechanical stress and wear on the diesel engine is minimized.

#For any start of a diesel generator, the diesel must be loaded in accordance with the manufacturer's recommendations.

SURVEILLANCE REQUIREMENTS (Continued)

5. Verifying that on an ECCS actuation test signal, without loss of offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.

6. Simulating a loss of offsite power in conjunction with an ECCS actuation test signal, and:

    a) Verifying loss of power is detected and deenergization of the emergency busses and load shedding from the emergency busses.

    b) Verifying the diesel generator starts* on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds after receipt of the start signal, energizes the autoconnected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

7. Verifying that all automatic diesel generator trips, except engine overspeed, generator differential current, generator overcurrent, bus differential current and low lube oil pressure are automatically bypassed upon loss of voltage on the emergency bus concurrent with an ECCS actuation signal.#

8. Verifying the diesel generator operates for at least 24 hours. During the first 22 hours of this test, the diesel generator shall be loaded to between 4300 and 4400 kW** and during the remaining 2 hours of this test, the diesel generator shall be loaded to between 4800 and 4873 kW. The generator voltage and

---

*This diesel generator start (10 sec) and subsequent loading (130 sec) from ambient conditions may be preceded by an engine prelube period and/or other warmup procedures recommended by the manufacturer so that mechanical stress and wear on the diesel engine is minimized.

**This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band shall not invalidate the test; the loads; however, shall not be less than 4300 kW nor greater than 4873 kW.

#Generator differential current, generator overcurrent, and bus differential current is two-out-of-three logic and low lube oil pressure is two-out-of-four logic.

frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24-hour test, perform Surveillance Requirement 4.8.1.1.2.h.4.b).**

9. Verifying that the auto-connected loads to each diesel generator do not exceed the continuous rating of 4430 kW.

10. Verifying the diesel generator's capability to:

    a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

    b) Transfer its loads to the offsite power source,

    c) Be restored to its standby status, and

    d) Diesel generator circuit breaker is open.

11. Verifying that with the diesel generator operating in a test mode and connected to its bus, a simulated ECCS actuation signal overrides the test mode by (1) returning the diesel generator to standby operation, and (2) automatically energizes the emergency loads with offsite power.

12. Verifying that the fuel oil transfer pump transfers fuel oil from each fuel storage tank to the day tank of each diesel via the installed cross connection lines.

13. Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval.

14. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

    a) Engine overspeed, generator differential, and low lube oil pressure (regular lockout relay, (1) 86R).
    b) Backup generator differential and generator overcurrent (backup lockout relay, (1) 86B)
    c) Generator ground and lockout relays-regular, backup and test, energized (breaker failure lockout relay, (1) 86F)

**If Surveillance Requirement 4.8.1.1.2.h.4.b) is not satisfactorily completed, it is not necessary to repeat the preceding 24 hour test. Instead, the diesel generator may be operated at between 4300 kw and 4400 kw for one hour or until operating temperature has stabilized prior to repeating Surveillance Requirement 4.8.1.1.2.h.4.b).

SURVEILLANCE REQUIREMENTS (Continued)

i. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting all diesel generators simultaneously, during shutdown, and verifying that all diesel generators accelerate to at least 514 rpm in less than or equal to 10 seconds.

j. At least once per 10 years by:

  1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution or equivalent, and

  2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code in accordance with ASME Code Section XI Article IWD-5000.

4.8.1.1.3 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission within 30 days pursuant to Specification 6.9.2. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests, on a per nuclear unit basis, is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

TABLE 4.8.1.1.2-1

DIESEL GENERATOR TEST SCHEDULE

Number of Failures in Last 20 Valid Tests*	Number of Failures in Last 100 Valid Tests*	Test Frequency
≤ 1	≤ 4	Once per 31 days
≥ 2**	≥ 5	Once per 7 days

---

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, but determined on a per diesel generator basis.

For the purposes of determining the required test frequency, the previous test failure count may be reduced to zero if a complete diesel overhaul to like-new condition is completed, provided that the overhaul including appropriate post-maintenance operation and testing, is specifically approved by the manufacturer and if acceptable reliability has been demonstrated. The reliability criterion shall be the successful completion of 14 consecutive tests in a single series. Ten of these tests shall be in accordance with the routine Surveillance Requirement 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5, four tests, in accordance with the 184-day testing requirement of Surveillance Requirement 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5. If this criterion is not satisfied during the first series of tests, any alternate criterion to be used to transvalue the failure count to zero requires NRC approval.

**The associated test frequency shall be maintained until seven consecutive failure free demands have been performed and the number of failures in the last 20 valid demands has been reduced to one.

A.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two diesel generators, one of which shall be diesel generator A or diesel generator B, each with:

   1. A separate fuel oil day tank containing a minimum of 200 gallons of fuel.

   2. A fuel storage system consisting of two storage tanks containing a minimum of 48,800 gallons of fuel.

   3. A separate fuel transfer pump for each storage tank.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With less than the above required A.C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment, operations with a potential for draining the reactor vessel and crane operations over the spent fuel storage pool when fuel assemblies are stored therein. In addition, when in OPERATIONAL CONDITION 5 with the water level less than 22'-2" above the reactor pressure vessel flange, immediately initiate corrective action to restore the required power sources to OPERABLE status as soon as practical.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.1.2 At least the above required A.C. electrical power sources shall be demonstrated OPERABLE per Surveillance Requirements 4.8.1.1.1, 4.8.1.1.2, and 4.8.1.1.3, except for the requirement of 4.8.1.1.2.a.5.

---

*When handling irradiated fuel in the secondary containment.

BEST COPY AVAILABLE

3/4.8.2 D.C. SOURCES

D.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical power sources shall be OPERABLE:

a. Channel A, consisting of:
   1. 125 volt battery 1AD411
   2. 125 volt full capacity charger 1AD413 or 1AD414
   3. 250 volt battery 10D421;
   4. 250 volt full capacity charger 10D423

b. Channel B, consisting of:
   1. 125 volt battery 1BD411
   2. 125 volt full capacity charger 1BD413 or 1BD414
   3. 250 volt battery 10D431;
   4. 250 volt full capacity charger 10D433

c. Channel C, consisting of:
   1. 125 volt battery 1CD411
   2. 125 volt full capacity charger 1CD413 or 1CD414
   3. 125 volt battery 1CD447
   4. 125 volt full capacity charger 1CD444

d. Channel D, consisting of:
   1. 125 volt battery 1DD411
   2. 125 volt full capacity charger 1DD413 or 1DD414
   3. 125 volt battery 1DD447
   4. 125 volt full capacity charger 1DD444

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With any 125v battery and/or all associated chargers of the above required D.C. electrical power sources inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With any 250v battery and/or charger of the above required DC electrical power sources inoperable, declare the associated HPCI or RCIC system inoperable and take the appropriate ACTION required by the applicable Specification.

300

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each of the above required batteries and chargers shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

   1. The parameters in Table 4.8.2.1-1 meet the Category A limits, and
   2. Total battery terminal voltage for each 125-volt battery is greater than or equal to 129 volts on float charge and for each 250-volt battery the terminal voltage is greater than or equal to 258 volts on float charge.

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 105 volts for a 125-volt battery or 210 volts for a 250-volt battery, or battery overcharge with battery terminal voltage above 140 volts for a 125-volt battery or 280 volts for a 250-volt battery, by verifying that:

   1. The parameters in Table 4.8.2.1-1 meet the Category B limits,
   2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohms, excluding cable intercell connections, and
   3. The average electrolyte temperature of each sixth cell of connected cells is above 60°F.

c. At least once per 18 months by verifying that:

   1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,
   2. The cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material,
   3. The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohms, excluding cable intercell connections, and
   4. The battery charger will supply the current listed below at the voltage listed below for at least 8 hours.*

CHARGER	Minimum Voltage	CURRENT (AMPERES)
1AD413, 1AD414 1BD413, 1BD414 1CD413, 1CD414 1CD444, 1DD414 1DD444, 1DD413	129	200
10D423, 10D433	258	50

---

*Prior to startup following the first refueling outage, this test may be performed for at least 4 hours.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months, during shutdown, by verifying that either:

1. The battery capacity is adequate to supply and maintain in OPERABLE status all of the actual emergency loads for the design duty cycle when the battery is subjected to a battery service test, or

2. The battery capacity is adequate to supply a dummy load of the following design profile while maintaining the battery terminal voltage greater than or equal to 105 volts for the 125-volt battery and 210 volts for the 250-volt battery:

Load Profile

125 Volt Battery	Amperes	Duration in Sequence (Minutes)
1AD411	451.1	1
	330.4	1
	273.0	8
	270.4	50
	298.1	180
1BD411	443.3	1
	326.5	1
	267.2	8
	267.9	30
	267.2	20
	289.6	180
1CD411	419.2	1
	359.2	1
	272.2	58
	299.2	180
1DD411	416.3	1
	356.3	1
	269.3	58
	294.3	180
125 Volt Battery		
1CD447	68	60
	77	180
1DD447	73	60
	80	180
250 Volt Battery		
10D421	758.1	1
	42.6	7

Load Profile (Continued)

250 Volt Battery	Amperes	Duration in Sequence (Minutes)
10D421 (Continued)		
	307.9	1
	42.6	41
	348.9	1
	42.6	7
	307.9	1
	42.6	41
	348.9	1
	42.6	7
	307.9	1
	42.6	41
	387.9	1
	83.6	89
10D431	197.9	1
	25.3	5
	66.3	1
	25.3	17
	74.7	1
	33.7	125
	33.7	60
	56.3	1
	33.7	29

e. At least once per 60 months during shutdown by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. At this once per 60 month interval, this performance discharge test may be performed in lieu of the battery service test.

f. At least once per 18 months during shutdown performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating. At this once per 18 months interval, this performance discharge test may be performed in lieu of the battery service test.

TABLE 4.8.2.1-1

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A(1)	CATEGORY B(2)	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable(3) value for each connected cell
Electrolyte Level	$\geq$Minimum level indication mark, and < ¼" above maximum level indication mark(d)	$\geq$Minimum level indication mark, and $\leq$ ¼" above maximum level indication mark(d)	Above top of plates, and not overflowing
Float Voltage	$\geq$ 2.13 volts	$\geq$ 2.13 volts(c) $\geq$ 1.195	> 2.07 volts Not more than .020 below the average of all connected cells
Specific Gravity(a)	$\geq$ 1.200(b)	Average of all connected cells > 1.205	Average of all connected cells $\geq$ 1.195(b)

(a) Corrected for electrolyte temperature and level.

(b) Or battery charging current is less than 2 amperes when on float charge.

(c) May be corrected for average electrolyte temperature.

(d) Electrolyte level may exceed 1/4" above maximum level indication mark if an equalizing charge is in progress or an equalizing charge has been completed within the previous 72 hours.

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.

(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

D.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, two of the following four channels of the D.C. electrical power sources, one of which shall be channel A or channel B, shall be OPERABLE with:

a. Channel A, consisting of:
  1. 125 volt battery 1AD411
  2. 125 volt full capacity charger# 1AD413 or 1AD414

b. Channel B, consisting of:
  1. 125 volt battery 1BD411
  2. 125 volt full capacity charger# 1BD413 or 1BD414.

c. Channel C, consisting of:
  1. 125 volt battery 1CD411
  2. 125 volt full capacity charger# 1CD413 or 1CD414
  3. 125 volt battery 1CD447
  4. 125 volt full capacity charger 1CD444

d. Channel D, consisting of:
  1. 125 volt battery 1DD411
  2. 125 volt full capacity charger# 1DD413 or 1DD414
  3. 125 volt battery 1DD447
  4. 125 volt full capacity charger 1DD444

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With less than two channels of the above required D.C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 At least the above required battery and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

---

*When handling irradiated fuel in the secondary containment.

#Only one full capacity charger per battery is required for the channel to be OPERABLE.

3/4.8.3 ONSITE POWER DISTRIBUTION SYSTEMS

DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following power distribution system channels shall be energized:

a. A.C. power distribution:

1. Channel A, consisting of:

a) 4160 volt A.C. switchgear bus	10A401
b) 480 volt A.C. load centers	10B410
	10B450
c) 480 volt A.C. MCCs	10B212
	10B411
	10B451
	10B553
d) 208/120 volt A.C. distribution panels	10Y401(source:10B411)
	10Y411(source:10B451)
	10Y501(source:10B553)
e) 120 volt A.C. distribution panels	1AJ481
	1YF401(source:1AJ481)
	1AJ482

2. Channel B, consisting of:

a) 4160 volt A.C. switchgear bus	10A402
b) 480 volt A.C. load centers	10B420
	10B460
c) 480 volt A.C. MCCs	10B222
	10B421
	10B461
	10B563
d) 208/120 volt A.C. distribution panels	10Y402(source:10B421)
	10Y412(source:10B461)
	10Y502(source:10B563)
e) 120 volt A.C. distribution panels	1BJ481
	1YF402(source:1BJ481)
	1BJ482

3. Channel C, consisting of:

a) 4160 volt A.C. switchgear bus	10A403
b) 480 volt A.C. load centers	10B430
	10B470
c) 480 volt A.C. MCCs	10B232
	10B431
	10B471
	10B573
d) 208/120 volt A.C. distribution panels	10Y403(source:10B431)
	10Y413(source:10B471)
	10Y503(source:10B573)

e) 120 volt A.C. distribution panels	1CJ481
	1YF403(source:1CJ481)
	1CJ482
4. Channel D, consisting of:	
a) 4160 volt A.C. switchgear bus	10A404
b) 480 volt A.C. load centers	10B440
	10B480
c) 480 volt A.C. MCCs	10B242
	10B441
	10B481
	10B583
d) 208/120 volt A.C. distribution panels	10Y404(source:10B441)
	10Y414(source:10B481)
	10Y504(source:10B583)
e) 120 volt A.C distribution panels	1DJ481
	1YF404(source:1DJ481)
	1DJ482

b. D.C. power distribution:

1. Channel A, consisting of:	
a) 125 volt D.C. switchgear	10D410
b) 125 volt D.C. fuse box	1AD412
c) 125 volt D.C. distribution panel	1AD417
d) 250 volt D.C. switchgear	10D450
e) 250 volt D.C. fuse box	10D422
f) 250 volt D.C. MCC	10D251
2. Channel B, consisting of:	
a) 125 volt D.C. switchgear	10D420
b) 125 volt D.C. fuse box	1BD412
c) 125 volt D.C. distribution panel	1BD417
d) 250 volt D.C. switchgear	10D460
e) 250 volt D.C. fuse boxes	10D432
f) 250 volt D.C. MCC	10D261
3. Channel C, consisting of:	
a) 125 volt D.C. switchgear	10D430
	10D436
b) 125 volt D.C. fuse box	1CD412
	1CD448
c) 125 volt D.C. distribution panel	1CD417
4. Channel D, consisting of:	
a) 125 volt D.C. switchgear	10D440
	10D446
b) 125 volt D.C. fuse boxes	1DD412
	1DD448
c) 125 volt D.C. distribution panel	1DD417

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one of the above required A.C. distribution system channels not energized, re-energize the channel within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one of the above required 125 volt D.C. distribution system channels not energized, re-energize the division within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any one of the above required 250 volt D.C. distribution systems not energized, declare the associated HPCI or RCIC system inoperable and apply the appropriate ACTION required by the applicable Specifications.

SURVEILLANCE REQUIREMENTS

4.8.3.1 Each of the above required power distribution system channels shall be determined energized at least once per 7 days by verifying correct breaker/switch alignment and voltage on the busses/MCCs/panels.

DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, 2 of the 4 channels, one of which shall be channel A or channel B, of the power distribution system shall be energized with:

a. A.C. power distribution:

1. Channel A, consisting of:

a) 4160 volt A.C. switchgear bus	10A401
b) 480 volt A.C. load centers	10B410
	10B450
c) 480 volt A.C. MCCs	10B212
	10B411
	10B451
	10B553
d) 208/120 volt A.C. distribution panels	10Y401(source:10B411)
	10Y411(source:10B451)
	10Y501(source:10B553)
e) 120 volt A.C. distribution panels	1AJ481
	1YF401(source:1AJ481)
	1AJ482

2. Channel B, consisting of:

a) 4160 volt A.C. switchgear bus	10A402
b) 480 volt A.C. load centers	10B420
	10B460
c) 480 volt A.C. MCCs	10B222
	10B421
	10B461
	10B563
d) 208/120 volt A.C. distribution panels	10Y402(source:10B421)
	10Y412(source:10B461)
	10Y502(source:10B563)
e) 120 volt A.C. distribution panels	1BJ481
	1YF402(source:1BJ481)
	1BJ482

3. Channel C, consisting of:

a) 4160 volt A.C. switchgear bus	10A403
b) 480 volt A.C. load centers	10B430
	10B470
c) 480 volt A.C. MCCs	10B232
	10B431
	10B471
	10B573
d) 208/120 volt A.C. distribution panels	10Y403(source:10B431)
	10Y413(source:10B471)
	10Y503(source:10B573)

e)	120 volt A.C. distribution panels	1CJ481
		1YF403(source:1CJ481)
		1CJ482
4.	Channel D, consisting of:	
a)	4160 volt A.C. switchgear bus	10A404
b)	480 volt A.C. load centers	10B440
		10B480
c)	480 volt A.C. MCCs	10B242
		10B441
		10B481
		10B583
d)	208/120 volt A.C. distribution panels	10Y404(source:10B441)
		10Y414(source:10B481)
		10Y504(source:10B583)
e)	120 volt A.C. distribution panels	1DJ481
		1YF404(source:1DJ481)
		1DJ482

b. D.C. power distribution:

1.	Channel A, consisting of:	
a)	125 volt D.C. switchgear	10D410
b)	125 volt D.C. fuse box	1AD412
c)	125 volt D.C. distribution panel	1AD417
2.	Channel B, consisting of:	
a)	125 volt D.C. switchgear	10D420
b)	125 volt D.C. fuse box	1BD412
c)	125 volt D.C. distribution panel	1BD417
3.	Channel C, consisting of:	
a)	125 volt D.C. switchgear	10D430
		10D436
b)	125 volt D.C. fuse boxes	1CD412
		1CD448
c)	125 volt D.C. distribution panel	1CD417
4.	Channel D, consisting of:	
a)	125 volt D.C. switchgear	10D440
		10D446
b)	125 volt D.C. fuse box	1DD412
		1DD448
c)	125 volt D.C. distribution panel	1DD417

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With less than two channels of the above required A.C. distribution system energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

b. With less than two channels of the above required D.C. distribution system energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.3.2 At least the above required power distribution system channels shall be determined energized at least once per 7 days by verifying correct breaker/switch alignment and voltage on the busses/MCCs/panels.

---

*When handling irradiated fuel in the secondary containment.

311

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 All primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.1-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more of the primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.1-1 inoperable, declare the affected system or component inoperable and apply the appropriate ACTION statement for the affected system, and

    1. For 4.16 kV circuit breakers, de-energize the 4.16 kV circuit(s) by tripping the associated redundant circuit breaker(s) within 72 hours and verify the redundant circuit breaker to be tripped at least once per 7 days thereafter.

    2. For 480 volt circuit breakers, remove the inoperable circuit breaker(s) from service by disconnecting* the breaker within 72 hours and verify the inoperable breaker(s) to be disconnected at least once per 7 days thereafter.

    Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. The provisions of Specification 3.0.4 are not applicable to overcurrent devices in 4.16 kV circuits which have their redundant circuit breakers tripped or to 480 volt circuits which have the inoperable circuit breaker disconnected.*

SURVEILLANCE REQUIREMENTS

4.8.4.1 Each of the primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.1-1 shall be demonstrated OPERABLE:

a. At least once per 18 months:

    1. By verifying that each of the medium voltage 4.16 kV circuit breakers are OPERABLE by performing:

        a) A CHANNEL CALIBRATION of the associated protective relays, and

        b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and overcurrent control circuits function as designed.

---

*After being disconnected, these breakers shall be maintained disconnected under administrative control.

SURVEILLANCE REQUIREMENTS (Continued)

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current with a value between 150% and 300% of the pickup of the long time delay trip element and verifying that the circuit breaker operates within the time delay bandwidth for that current specified by the manufacturer. The instantaneous element shall be tested by injecting a current in excess of 120% the pickup value of the element and verifying that the circuit breaker trips instantaneously with no intentional time delay. Molded case circuit breaker testing shall also follow this procedure except that generally no more than two trip elements, time delay and instantaneous, will be involved. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

TABLE 3.8.4.1-1

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

1. 4160-VOLT CIRCUIT BREAKERS

CIRCUIT BREAKER NO.	LOCATION	SYSTEMS OR EQUIPMENT POWERED
1AN205	1AN205	Reactor Recirculation Pump 1AP201
1BN205	1BN205	Reactor Recirculation Pump 1BP201
1CN205	1CN205	Reactor Recirculation Pump 1AP201
1DN205	1DN205	Reactor Recirculation Pump 1BP201

2. 480-VOLT MOLDED CASE CIRCUIT BREAKERS

Primary and backup breakers have the same device numbers and are located in the same Motor Control Center cubicle.

CIRCUIT BREAKER NO.	LOCATION	TYPES	SYSTEMS OR EQUIPMENT POWERED
52-411065	10B411	IM HFB150 TM HFB150	RHR Head Spray Valve 1BC-HV-F022
52-451061	10B451	IM HFB150 TM HFB150	RHR Shutdown Cooling Suction Inboard Valve 1BC-HV-F009
52-212021	10B212	IM HFB150 TM HFB150	RWCU Suction Isolation Inboard Valve 1BG-HV-F001
52-212101	10B212	IM HFB150 TM HFB150	PCIGS Drywell Supply Header A Isolation Valve 1KL-HV-5152A
52-212181	10B212	IM HFB150 TM HFB150	Main Steam Line Drain Inboard Valve 1AB-HV-F016
52-212183	10B212	IM HFB150 TM HFB150	PCIGS Drywell Suction Inboard Valve 1KL-HV-5148
52-232061	10B232	IM HFB150 TM HFB150	Drywell Supply Header A Isolation Valve 1KL-HV-5124A
52-232103	10B232	IM HFB150 TM HFB150	Drywell Equip. Drain Sump Isolation Valve 1HB-HV-F019
52-232104	10B232	IM HFB150 TM HFB150	HPCI Warmup Bypass Line Isolation Valve 1FD-HV-F100
52-232181	10B232	IM HFB150 TM HFB150	Chilled Water Loop A Supply Isolation Valve 1GB-HV-9531B1
52-232182	10B232	IM HFB150 TM HFB150	Chilled Water Loop A Return Isolation Valve 1GB-HV-9531B2
52-232183	10B232	IM HFB150 TM HFB150	Chilled Water Loop B Supply Isolation Valve 1GB-HV-9531B3

TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

2. 480-VOLT MOLDED CASE CIRCUIT BREAKERS (Continued)

CIRCUIT BREAKER NO.	LOCATION	TYPES	SYSTEMS OR EQUIPMENT POWERED
52-232193	10B232	IM HFB150 TM HFB150	Chilled Water Loop B Return Isolation Valve 1GB-HV-9531B4
52-232203	10B232	IM HFB150 TM HFB150	HPCI Turbine Steam Supply Isolation Valve 1FD-HV-F002
52-242021	10B242	IM HFB150 TM HFB150	Drywell Floor Drain Sump Isolation Valve 1HB-HV-F003
52-242061	10B242	IM HFB150 TM HFB150	Drywell Supply Header B Isolation Valve 1KL-HV-5124B
52-242101	10B242	IM HFB150 TM HFB150	PCIGS Drywell Supply Header B Isolation Valve 1KL-HV-5152B
52-242102	10B242	IM HFB150 TM HFB150	RCIC Turbine Steam Supply Isolation Valve 1FC-HV-F007
52-242103	10B242	IM HFB150 TM HFB150	RCIC Warmup Bypass Line Isolation Valve 1FC-HV-F076
52-242172	10B242	IM HFB150 TM HFB150	Reactor Recirc Pumps Cooling Supply Isolation 1ED-HV-2554
52-242173	10B242	IM HFB150 TM HFB150	Reactor Recirc Pumps Cooling Return Isolation 1ED-HV-2556
52-252021	10B252	IM HFB150 TM HFB150	Drywell Cooler A Fan 1A1V212
52-252022	10B252	IM HFB150 TM HFB150	Drywell Cooler B Fan 1B1V212
52-252031	10B252	IM HFB150 TM HFB150	Drywell Cooler C Fan 1C1V212
52-252032	10B252	IM HFB150 TM HFB150	Drywell Cooler D Fan 1D1V212
52-252041	10B252	IM HFB150 TM HFB150	Drywell Cooler E Fan 1E1V212
52-252042	10B252	IM HFB150 TM HFB150	Drywell Cooler F Fan 1F1V212
52-252051	10B252	IM HFB150 TM HFB150	Drywell Cooler G Fan 1G1V212
52-252052	10B252	IM HFB150 TM HFB150	Drywell Cooler H Fan 1H1V212

315

TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

2. 480-VOLT MOLDED CASE CIRCUIT BREAKERS (Continued)

CIRCUIT BREAKER NO.	LOCATION	TYPES	SYSTEMS OR EQUIPMENT POWERED
52-252063	10B252	IM HFB150 TM HFB150	Drywell Equip Drain Sump Pump 1AP267
52-252064	10B252	IM HFB150 TM HFB150	Drywell Floor Drain Sump Pump 1CP267
52-252073	10B252	IM HFB150 TM HFB150	Feedwater Inlet A Shutoff 1AE-HV-F011A
52-262021	10B262	IM HFB150 TM HFB150	Drywell Cooler A Fan 1A2V212
52-262022	10B262	IM HFB150 TM HFB150	Drywell Cooler B Fan 1B2V212
52-262031	10B262	IM HFB150 TM HFB150	Drywell Cooler C Fan 1C2V212
52-262032	10B262	IM HFB150 TM HFB150	Drywell Cooler D Fan 1D2V212
52-262041	10B262	IM HFB150 TM HFB150	Drywell Cooler E Fan 1E2V212
52-262042	10B262	IM HFB150 TM HFB150	Drywell Cooler F Fan 1F2V212
52-262051	10B262	IM HFB150 TM HFB150	Drywell Cooler G Fan 1G2V212
52-262052	10B262	IM HFB150 TM HFB150	Drywell Cooler H Fan 1H2V212
52-262063	10B262	IM HFB150 TM HFB150	Drywell Equip Drain Sump Pump 1BP267
52-262064	10B262	IM HFB150 TM HFB150	Drywell Floor Drain Sump Pump 1DP267
52-253012*	10B253	IM HFB150 TM HFB150	Recirc Pump Motor Hoist 1AH201 Disconnect Switch 1AS204
52-253021	10B253	IM HFB150 TM HFB150	Recirc Pump 1BP201 Suction Valve 1BB-HV-F023B
52-253031	10B253	IM HFB150 TM HFB150	Recirc Pump 1BP201 Discharge Valve 1BB-HV-F031B
52-253053	10B253	IM HFB150 TM HFB150	Reactor Vessel Head Vent Inboard Isolation 1BB-HV-F001

TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

2. 480-VOLT MOLDED CASE CIRCUIT BREAKERS (Continued)

CIRCUIT BREAKER NO.	LOCATION	TYPES	SYSTEMS OR EQUIPMENT POWERED
52-253064	10B253	IM HFB150 TM HFB150	Reactor Vessel Head Vent to Steam Line 1BB-HV-F005
52-263011	10B263	IM HFB150 TM HFB150	Reactor Vessel Head Vent Outboard Isolation 1BB-HV-F002
52-263012*	10B263	IM HFB150 TM HFB150	Recirc Pump Motor Hoist 1BH201 Disconnect Switch 1BS204
52-263022*	10B263	TM HFB150	CRD Equipment Handling Platform 10S270
52-263042*	10B263	IM HFB150 TM HFB150	Main Steam Relief Valve Hoist 10H202 Disconnect Switch 10S207
52-263054	10B263	IM HFB150 TM HFB150	RWCU Suction from Recirc Loop A 1BG-HV-F100
52-263081	10B263	IM HFB150 TM HFB150	RWCU Suction from RPV Drain Valve 1BG-HV-F101
52-263082	10B263	IM HFB150 TM HFB150	RWCU Suction Valve 1BG-HV-F102
52-263083	10B263	IM HFB150 TM HFB150	RWCU Suction from Recirc Loop B Valve 1BG-HV-F106
52-264053	10B264	IM HFB150 TM HFB150	Recirc Pump A Discharge Valve 1BB-HV-F031A
52-264062	10B264	IM HFB150 TM HFB150	Feedwater Inlet B Shutoff Valve 1AE-HV-F011B
52-264071	10B264	IM HFB150 TM HFB150	Reactor Recirc Pump 1AP201 Space Heater 1AS220
52-264072	10B264	IM HFB150 TM HFB150	Reactor Recirc Pump 1BP201 Space Heater 1BS220
52-264083	10B264	IM HFB150 TM HFB150	Recirc Pump A Suction Valve 1BB-HV-F023A

---

*These breakers shall be administratively maintained open in OPERATIONAL CONDITIONS 1, 2 and 3 and are not required to be tested.

317

MOTOR OPERATED VALVES - THERMAL OVERLOAD PROTECTION (BYPASSED)

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection bypass circuit of each motor operated valve (MOV) shown in Table 3.8.4.2-1 shall be OPERABLE.

APPLICABILITY: Whenever the MOV is required to be OPERABLE.

ACTION:

With the thermal overload protection bypass circuit for one or more of the above required MOVs inoperable, restore the inoperable thermal overload protection bypass circuit(s) to OPERABLE status within 8 hours or declare the affected MOV(s) inoperable and apply the appropriate ACTION statement(s) for the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2.1 The thermal overload protection bypass circuit for each of the above required MOVs shall be demonstrated OPERABLE:

a. At least once per 18 months by the performance of a CHANNEL FUNCTIONAL TEST for:

    1. Those thermal overload protection devices which are normally in force during plant operation and bypassed only under accident conditions.

    2. A representative sample of at least 25% of those thermal overload protection devices which are bypassed continuously and temporarily placed in force only when the MOVs are undergoing periodic or maintenance testing, such that the bypass circuitry for each thermal overload protection device of this type is tested at least once per 6 years.

    3. A representative sample of at least 25% of those thermal overload protection devices which are in force during normal manual (momentary push button contact) MOV operation and bypassed during remote manual (push button held depressed) MOV operation, such that the bypass circuitry for each thermal overload protection device of this type is tested at least once per 6 years.

b. Following maintenance on the motor starter.

4.8.4.2.2 The thermal overload protection for the above required MOVs which are continuously bypassed and temporarily placed in force only when the MOV is undergoing periodic or maintenance testing shall be verified to be continuously bypassed following such testing.

TABLE 3.8.4.2-1

MOTOR OPERATED VALVES - THERMAL OVERLOAD PROTECTION (BYPASSED)

VALVE NUMBER	THERMAL OVERLOAD PROTECTION STATUS	SYSTEM(S) AFFECTED
1AB-HV-F016	2	Main Steam
1AB-HV-F019	2	Main Steam
1AB-HV-F067A	2	Main Steam
1AB-HV-F067B	2	Main Steam
1AB-HV-F067C	2	Main Steam
1AB-HV-F067D	2	Main Steam
1AB-HV-F071	3	Main Steam
1AB-HV-3631A	3	Main Steam
1AB-HV-3631B	3	Main Steam
1AB-HV-3631C	3	Main Steam
1AB-HV-3631D	3	Main Steam
1AE-HV-F032A	3	Feedwater
1AE-HV-F032B	3	Feedwater
1AE-HV-F039	3	Feedwater
1AE-HV-4144	3	Feedwater
1AN-HV-2600	3	Demineralized Water
0AP-HV-2072	3	Condensate Storage & Transfer
0AP-HV-2073	3	Condensate Storage & Transfer
1AP-HV-F011	1	Condensate Storage & Transfer
1BC-HV-F004A	3	Residual Heat Removal (RHR)
1BC-HV-F004B	3	RHR
1BC-HV-F004C	3	RHR
1BC-HV-F004D	3	RHR
1BC-HV-F006A	3	RHR
1BC-HV-F006B	3	RHR
1BC-HV-F007A	1	RHR
1BC-HV-F007B	1	RHR
1BC-HV-F007C	1	RHR
1BC-HV-F007D	1	RHR
1BC-HV-F008	2	RHR
1BC-HV-F009	2	RHR
1BC-HV-F010A	2	RHR
1BC-HV-F010B	2	RHR
1BC-HV-F015A	2	RHR
1BC-HV-F015B	2	RHR
1BC-HV-F016A	3	RHR
1BC-HV-F016B	3	RHR
1BC-HV-5017A	1	RHR
1BC-HV-5017B	1	RHR
1BC-HV-F017C	1	RHR
1BC-HV-F017D	1	RHR
1BC-HV-F021A	3	RHR
1BC-HV-F021B	3	RHR
1BC-HV-F022	2	RHR
1BC-HV-F023	2	RHR

TABLE 3.8.4.2-1 (Continued)

MOTOR OPERATED VALVES - THERMAL OVERLOAD PROTECTION (BYPASSED)

VALVE NUMBER	THERMAL OVERLOAD PROTECTION STATUS	SYSTEM(S) AFFECTED
1BC-HV-F024A	2	RHR
1BC-HV-F024B	2	RHR
1BC-HV-F027A	2	RHR
1BC-HV-F027B	2	RHR
1BC-HV-F040	2	RHR
1BC-HV-F047A	3	RHR
1BC-HV-F047B	3	RHR
1BC-HV-F048A	1	RHR
1BC-HV-F048B	1	RHR
1BC-HV-F049	2	RHR
1BC-HV-F075	3	RHR
1BC-HV-4439	3	RHR
1BC-HV-5055A	2	Containment Atmosphere Control
1BC-HV-5055B	2	Containment Atmosphere Control
1BD-HV-F010	2	Reactor Core Isolation Cooling (RCIC)
1BD-HV-F012	1	RCIC
1BD-HV-F013	1	RCIC
1BD-HV-F022	2	RCIC
1BD-HV-F031	1	RCIC
1BD-HV-F046	1	RCIC
1BE-HV-F001A	3	Core Spray
1BE-HV-F001B	3	Core Spray
1BE-HV-F001C	3	Core Spray
1BE-HV-F001D	3	Core Spray
1BE-HV-F004A	1	Core Spray
1BE-HV-F004B	1	Core Spray
1BE-HV-F005A	1	Core Spray
1BE-HV-F005B	1	Core Spray
1BE-HV-F015A	2	Core Spray
1BE-HV-F015B	2	Core Spray
1BE-HV-F031A	1	Core Spray
1BE-HV-F031B	1	Core Spray
1BF-HV-3800A	3	Control Rod Drive
1BF-HV-3800B	3	Control Rod Drive
1BF-HV-4005	3	Control Rod Drive
1BG-HV-F001	2	Reactor Water Cleanup
1BG-HV-F004	2	Reactor Water Cleanup
1BG-HV-F034	3	Reactor Water Cleanup
1BG-HV-F035	3	Reactor Water Cleanup
1BG-HV-3980	3	Reactor Water Cleanup
1BH-HV-F006A	3	Standby Liquid Control
1BH-HV-F006B	3	Standby Liquid Control

320

TABLE 3.8.4.2-1 (Continued)

MOTOR OPERATED VALVES - THERMAL OVERLOAD PROTECTION (BYPASSED)

VALVE NUMBER	THERMAL OVERLOAD PROTECTION STATUS	SYSTEM(S) AFFECTED
1BJ-HV-F004	1	High Pressure Coolant Injection (HPCI)
1BJ-HV-F006	1	HPCI
1BJ-HV-F007	1	HPCI
1BJ-HV-F008	2	HPCI
1BJ-HV-F012	1	HPCI
1BJ-HV-F042	1	HPCI
1BJ-HV-F059	1	HPCI
1BJ-HV-4803	3	HPCI
1BJ-HV-4804	3	HPCI
1BJ-HV-4865	3	HPCI
1BJ-HV-4866	3	HPCI
1BJ-HV-8278	1	HPCI
0BN-HV-2069	3	Refueling Water Transfer
1EA-HV-F073	3	Station Service Water
1EA-HV-2197A	3	Station Service Water
1EA-HV-2197B	3	Station Service Water
1EA-HV-2197C	3	Station Service Water
1EA-HV-2197D	3	Station Service Water
1EA-HV-2198A	2	Station Service Water
1EA-HV-2198B	2	Station Service Water
1EA-HV-2198C	2	Station Service Water
1EA-HV-2198D	2	Station Service Water
1EA-HV-2203	3	Station Service Water
1EA-HV-2204	3	Station Service Water
1EA-HV-2207	3	Station Service Water
1EA-HV-2234	3	Station Service Water
1EA-HV-2236	3	Station Service Water
1EA-HV-2238	3	Station Service Water
1EA-HV-2346	3	Station Service Water
1EA-HV-2355A	2	Station Service Water
1EA-HV-2355B	2	Station Service Water
1EA-HV-2356A	3	Station Service Water
1EA-HV-2356B	3	Station Service Water
1EA-HV-2357A	3	Station Service Water
1EA-HV-2357B	3	Station Service Water
1EA-HV-2371A	2	Station Service Water
1EA-HV-2371B	2	Station Service Water
1EC-HV-4647	3	Fuel Pool Cooling
1EC-HV-4648	3	Fuel Pool Cooling
1EC-HV-4689A	3	Fuel Pool Cooling
1EC-HV-4689B	3	Fuel Pool Cooling
1ED-HV-2553	2	Reactor Auxiliaries Cooling
1ED-HV-2554	2	Reactor Auxiliaries Cooling

TABLE 3.8.4.2-1 (Continued)

MOTOR OPERATED VALVES - THERMAL OVERLOAD PROTECTION (BYPASSED)

VALVE NUMBER	THERMAL OVERLOAD PROTECTION STATUS	SYSTEM(S) AFFECTED
1ED-HV-2555	2	Reactor Auxiliaries Cooling
1ED-HV-2556	2	Reactor Auxiliaries Cooling
1ED-HV-2598	3	Reactor Auxiliaries Cooling
1ED-HV-2599	3	Reactor Auxiliaries Cooling
1EE-HV-4652	2	Torus Water Cleanup
1EE-HV-4679	2	Torus Water Cleanup
1EE-HV-4680	2	Torus Water Cleanup
1EE-HV-4681	2	Torus Water Cleanup
1EG-HV-2314A	3	Safety Auxiliaries Cooling
1EG-HV-2314B	3	Safety Auxiliaries Cooling
1EG-HV-2317A	3	Safety Auxiliaries Cooling
1EG-HV-2317B	3	Safety Auxiliaries Cooling
1EG-HV-2320A	3	Safety Auxiliaries Cooling
1EG-HV-2320B	3	Safety Auxiliaries Cooling
1EG-HV-2321A	2	Safety Auxiliaries Cooling
1EG-HV-2321B	2	Safety Auxiliaries Cooling
1EG-HV-2446	3	Safety Auxiliaries Cooling
1EG-HV-2447	3	Safety Auxiliaries Cooling
1EG-HV-2452A	3	Safety Auxiliaries Cooling
1EG-HV-2452B	3	Safety Auxiliaries Cooling
1EG-HV-2453A	2	Safety Auxiliaries Cooling
1EG-HV-2453B	2	Safety Auxiliaries Cooling
1EG-HV-2491A	3	Safety Auxiliaries Cooling
1EG-HV-2491B	3	Safety Auxiliaries Cooling
1EG-HV-2494A	3	Safety Auxiliaries Cooling
1EG-HV-2494B	3	Safety Auxiliaries Cooling
1EG-HV-2496A	3	Safety Auxiliaries Cooling
1EG-HV-2496B	3	Safety Auxiliaries Cooling
1EG-HV-2496C	3	Safety Auxiliaries Cooling
1EG-HV-2496D	3	Safety Auxiliaries Cooling
1EG-HV-2512A	3	Safety Auxiliaries Cooling
1EG-HV-2512B	3	Safety Auxiliaries Cooling
1EG-HV-7921A	3	Safety Auxiliaries Cooling
1EG-HV-7921B	3	Safety Auxiliaries Cooling
1EG-HV-7922A	3	Safety Auxiliaries Cooling
1EG-HV-7922B	3	Safety Auxiliaries Cooling
1EP-HV-2225A	3	Station Service Water
1EP-HV-2225B	3	Station Service Water
1EP-HV-2225C	3	Station Service Water
1EP-HV-2225D	3	Station Service Water
1FC-HV-F007	2	Reactor Core Isolation Cooling (RCIC)
1FC-HV-F008	2	RCIC

322

TABLE 3.8.4.2-1 (Continued)

MOTOR OPERATED VALVES - THERMAL OVERLOAD PROTECTION (BYPASSED)

VALVE NUMBER	THERMAL OVERLOAD PROTECTION STATUS	SYSTEM(S) AFFECTED
1FC-HV-F045	2	RCIC
1FC-HV-F059	3	RCIC
1FC-HV-F060	3	RCIC
1FC-HV-F062	2	RCIC
1FC-HV-F076	2	RCIC
1FC-HV-F084	2	RCIC
1FC-HV-4282	3	RCIC
1FD-HV-F001	1	High Pressure Coolant Injection (HPCI)
1FD-HV-F002	2	HPCI
1FD-HV-F003	2	HPCI
1FD-HV-F071	3	HPCI
1FD-HV-F075	2	HPCI
1FD-HV-F079	2	HPCI
1FD-HV-F100	2	HPCI
1FD-HV-4922	2	HPCI
1GB-HV-9531A1	2	Chilled Water
1GB-HV-9531A2	2	Chilled Water
1GB-HV-9531A3	2	Chilled Water
1GB-HV-9531A4	2	Chilled Water
1GB-HV-9531B1	2	Chilled Water
1GB-HV-9531B2	2	Chilled Water
1GB-HV-9531B3	2	Chilled Water
1GB-HV-9531B4	2	Chilled Water
1GB-HV-9532-1	3	Chilled Water
1GB-HV-9532-2	3	Chilled Water
1GH-HV-5543	3	Radwaste Area Vent
1GS-HV-4955A	2	Containment Atmosphere Control
1GS-HV-4955B	2	Containment Atmosphere Control
1GS-HV-4959A	2	Containment Atmosphere Control
1GS-HV-4959B	2	Containment Atmosphere Control
1GS-HV-4965A	2	Containment Atmosphere Control
1GS-HV-4965B	2	Containment Atmosphere Control
1GS-HV-4966A	2	Containment Atmosphere Control
1GS-HV-4966B	2	Containment Atmosphere Control
1GS-HV-4974	2	Containment Atmosphere Control
1GS-HV-4983A	2	Containment Atmosphere Control
1GS-HV-4983B	2	Containment Atmosphere Control
1GS-HV-4984A	2	Containment Atmosphere Control
1GS-HV-4984B	2	Containment Atmosphere Control
1GS-HV-5019A	2	Containment Atmosphere Control
1GS-HV-5019B	2	Containment Atmosphere Control
1GS-HV-5022A	2	Containment Atmosphere Control
1GS-HV-5022B	2	Containment Atmosphere Control

TABLE 3.8.4.2-1 (Continued)

MOTOR OPERATED VALVES - THERMAL OVERLOAD PROTECTION (BYPASSED)

VALVE NUMBER	THERMAL OVERLOAD PROTECTION STATUS	SYSTEM(S) AFFECTED
1GS-HV-5050A	2	Containment Atmosphere Control
1GS-HV-5050B	2	Containment Atmosphere Control
1GS-HV-5052A	2	Containment Atmosphere Control
1GS-HV-5052B	2	Containment Atmosphere Control
1GS-HV-5053A	2	Containment Atmosphere Control
1GS-HV-5053B	2	Containment Atmosphere Control
1GS-HV-5054A	2	Containment Atmosphere Control
1GS-HV-5054B	2	Containment Atmosphere Control
1GS-HV-5057A	2	Containment Atmosphere Control
1GS-HV-5057B	2	Containment Atmosphere Control
1HB-HV-F003	2	Liquid Radwaste
1HB-HV-F004	2	Liquid Radwaste
1HB-HV-F019	2	Liquid Radwaste
1HB-HV-F020	2	Liquid Radwaste
1HB-HV-5262	3	Liquid Radwaste
1HB-HV-5275	3	Liquid Radwaste
1HC-HV-5551	3	Solid Radwaste
1KA-HV-7626	3	Service Compressed Air
1KB-HV-7629	3	Instrument Air (Backup to PCIG System)
1KL-HV-5124A	2	Primary Containment Instrument Gas (PCIG)
1KL-HV-5124B	2	PCIG
1KL-HV-5126A	2	PCIG
1KL-HV-5126B	2	PCIG
1KL-HV-5147	2	PCIG
1KL-HV-5148	2	PCIG
1KL-HV-5152A	2	PCIG
1KL-HV-5152B	2	PCIG
1KL-HV-5160A	3	PCIG
1KL-HV-5160B	3	PCIG
1KL-HV-5162	2	PCIG
1KL-HV-5172A	2	PCIG
1KL-HV-5172B	2	PCIG
1KP-HV-5829A	3	Main Steam Isolation Valve Sealing
1KP-HV-5829B	3	Main Steam Isolation Valve Sealing
1KP-HV-5834A	2	Main Steam Isolation Valve Sealing
1KP-HV-5834B	3	Main Steam Isolation Valve Sealing
1KP-HV-5835A	2	Main Steam Isolation Valve Sealing
1KP-HV-5835B	3	Main Steam Isolation Valve Sealing
1KP-HV-5836A	2	Main Steam Isolation Valve Sealing
1KP-HV-5836B	3	Main Steam Isolation Valve Sealing

BEST COPY A

TABLE 3.8.4.2-1 (Continued)

MOTOR OPERATED VALVES - THERMAL OVERLOAD PROTECTION (BYPASSED)

VALVE NUMBER	THERMAL OVERLOAD PROTECTION STATUS	SYSTEM(S) AFFECTED
1KP-HV-5837A	2	Main Steam Isolation Valve Sealing
1KP-HV-5837B	3	Main Steam Isolation Valve Sealing
1SK-HV-4953	2	Plant Leak Detection
1SK-HV-4957	2	Plant Leak Detection
1SK-HV-4981	2	Plant Leak Detection
1SK-HV-5018	2	Plant Leak Detection

THERMAL OVERLOAD PROTECTION STATUS CODES

1. Normally in force during plant operation and bypassed only under accident conditions.

2. Bypassed continuously and temporarily placed in force only when the MOVs are undergoing periodic or maintenance testing.

3. In force during normal remote manual (momentary push button contact) MOV operation and bypassed during remote manual (push button held depressed) MOV operation.

325

MOTOR OPERATED VALVES - THERMAL OVERLOAD PROTECTION (NOT BYPASSED)

LIMITING CONDITION FOR OPERATION

3.8.4.3 The thermal overload protection of each motor operated valve (MOV) shown in Table 3.8.4.3-1 shall be OPERABLE.

APPLICABILITY: Whenever the MOV is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required MOVs inoperable, restore the inoperable thermal overload(s) to OPERABLE status within 8 hours or declare the affected MOV(s) inoperable and apply the appropriate ACTION statement(s) for the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.3 The thermal overload protection for each of the above required MOVs shall be demonstrated OPERABLE at least once per 18 months and following maintenance on the motor starter by the performance of a CHANNEL CALIBRATION.

TABLE 3.8.4.3-1

MOTOR OPERATED VALVES - THERMAL OVERLOAD PROTECTION (NOT BYPASSED)

VALVE NUMBER	SYSTEM(S) AFFECTED
1BC-HV-F003A	Residual Heat Removal
1BC-HV-F003B	Residual Heat Removal
1GS-HV-5741A	Containment Atmosphere Control
1GS-HV-5741B	Containment Atmosphere Control
1KC-HV-3408M	Fire Protection

321

REACTOR PROTECTION SYSTEM ELECTRICAL POWER MONITORING

LIMITING CONDITION FOR OPERATION

3.8.4.4 Two RPS electric power monitoring channels for each inservice RPS MG set or alternate power supply shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one RPS electric power monitoring channel for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable power monitoring channel to OPERABLE status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

b. With both RPS electric power monitoring channels for an inservice RPS MG set or alternate power supply inoperable, restore at least one electric power monitoring channel to OPERABLE status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

SURVEILLANCE REQUIREMENTS

4.8.4.4 The above specified RPS electric power monitoring channels shall be determined OPERABLE:

a. At least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST, and

b. At least once per 18 months by demonstrating the OPERABILITY of over-voltage, under-voltage, and under-frequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints.

   1. Over-voltage $\leq$ 132 VAC, (Bus A), 132 VAC (Bus B)

   2. Under-voltage $\geq$ 108 VAC, (Bus A), 108 VAC (Bus B)

   3. Under-frequency $\geq$ 57 Hz. (Bus A and Bus B)

CLASS 1E ISOLATION BREAKER OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.5 All Class 1E isolation breaker (tripped by a LOCA signal) overcurrent protective devices shown in Table 3.8.4.5-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more of the overcurrent protective devices shown in Table 3.8.4.5-1 inoperable, declare the affected isolation breaker inoperable and remove the inoperable circuit breaker(s) from service within 72 hours and verify the inoperable breaker(s) to be disconnected at least once per 7 days thereafter.

b. The provisions of Specification 3.0.4 are not applicable to overcurrent devices in 480 volt circuits which have the inoperable circuit breaker disconnected.

SURVEILLANCE REQUIREMENTS

4.8.4.5 Each of the Class 1E isolation breaker overcurrent protective devices shown in Table 3.8.4.5-1 shall be demonstrated OPERABLE:

a. At least once per 18 months:

By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current with a value between 150% and 300% of the pickup of the long time delay trip element and a value between 150% and 250% of the pickup of the short time delay, and verifying that the circuit breaker operates within the time delay band width for that current specified by the manufacturer. The instantaneous element shall be tested by injecting a current in excess of 120% of the pickup value of the element and verifying that the circuit breaker trips instantaneously with no intentional time delay. Molded case circuit breaker testing shall also follow this procedure except that generally no more than two trip elements, time delay and instantaneous, will be involved. For circuit breakers equipped with solid state trip devices, the func-tional testing may be performed with use of portable instruments de-signed to verify the time-current characteristics and pickup calibra-tion of the trip elements. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

329

TABLE 3.8.4.5-1

CLASS 1E ISOLATION BREAKER

OVERCURRENT PROTECTIVE DEVICES
(BREAKER TRIPPED BY A LOCA SIGNAL)

480 VAC POWER CIRCUIT BREAKERS

1. TYPE AKR-5A-30

Class 1E Circuit Breaker No.	Class 1E Bus	Non-Class 1E Load Description
52-41011	10B410	Reactor Auxiliaries Cooling System Pump 1AP209
52-41014	10B410	Radwaste and Service Area MCC 10B313
52-41024	10B410	Reactor Building Supply Air Handling Unit 1BVH300
52-42011	10B420	Reactor Auxiliaries Cooling System Pump 1BP209
52-42014	10B420	Radwaste and Service Area MCC 10B323
52-42024	10B420	Reactor Building Exhaust Fan 1BV301
52-43024	10B430	Reactor Building Supply Air Handling Unit 1CVH300
52-43014	10B430	Control Rod Drive Pump 1AP207
52-44014	10B440	Control Rod Drive Pump 1BP207
52-44024	10B440	Reactor Building Supply Air Handling Unit 1AVH300
52-44034	10B440	Radwaste Area Supply Fan 0BV316
52-45011	10B450	Reactor Area MCC 10B252
52-45014	10B450	Radwaste Area Exhaust Fan 0AV305
52-45024	10B450	Emergency Instrument Air Compressor 10K100

BEST COPY [illegible]

330

TABLE 3.8.4.5-1 (Continued)

480 VAC POWER CIRCUIT BREAKERS

1. Type AKR-5A-30 (Continued)

Class 1E Circuit Breaker No.	Class 1E Bus	Non-Class 1E Load Description
52-45034	10B450	Reactor Building Exhaust Fan 1CV301
52-46011	10B460	Reactor Area MCC 10B262
52-46014	10B460	Radwaste Area Exhaust Fan 0BV305
52-47011	10B470	Reactor Area MCC 10B272
52-47014	10B470	Radwaste Area Exhaust Fan 0CV305
52-47024	10B470	Radwaste Area Supply Fan 0AV316
52-47031	10B470	Technical Support Center MCC 00B474
52-48011	10B480	Reactor Area MCC 10B282
52-48024	10B480	Reactor Building Exhaust Fan 1AV301

480 VAC MOLDED CASE CIRCUIT BREAKERS

1. Type HFB150

Class 1E Circuit Breaker No.	Class 1E Bus	Non-Class 1E Load Description
52-441043	10B441	NSSS Computer Inverter 10D485
52-451023	10B451	Public Address System Inverter 10D496
52-471023	10B471	Security System Inverter 0AD495

331

POWER RANGE NEUTRON MONITORING SYSTEM ELECTRICAL POWER MONITORING

LIMITING CONDITION FOR OPERATION

3.8.4.6 The power range neutron monitoring system (NMS) electric power monitoring channels for each inservice power range NMS power supply shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one power range NMS electric power monitoring channel for an inservice power range NMS power supply inoperable, restore the inoperable power monitoring channel to OPERABLE status within 72 hours or deenergize the associated power range NMS power supply feeder circuit.

b. With both power range NMS electric power monitoring channels for an inservice power range NMS power supply inoperable, restore at least one electric power monitoring channel to OPERABLE status within 30 minutes or deenergize the associated power range NMS power supply feeder circuit.

SURVEILLANCE REQUIREMENTS

4.8.4.6 The above specified power range NMS electric power monitoring channels shall be determined OPERABLE:

a. At least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST, and

b. At least once per 18 months by demonstrating the OPERABILITY of over-voltage, under-voltage, and under-frequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints.

1. Over-voltage $\leq$ 132 VAC (BUS A), 132 VAC (BUS B)
2. Under-voltage $\geq$ 108 VAC (BUS A), 108 VAC (BUS B)
3. Under-frequency $\geq$ 57 Hz. -0, +2%

332

3/4.9 REFUELING OPERATIONS

3/4.9.1 REACTOR MODE SWITCH

LIMITING CONDITION FOR OPERATION

3.9.1 The reactor mode switch shall be OPERABLE and locked in the Shutdown or Refuel position. When the reactor mode switch is locked in the Refuel position:

a. A control rod shall not be withdrawn unless the Refuel position one-rod-out interlock is OPERABLE.

b. CORE ALTERATIONS shall not be performed using equipment associated with a Refuel position interlock unless at least the following associated Refuel position interlocks are OPERABLE for such equipment.

   1. All rods in.
   2. Refuel platform position.
   3. Refuel platform hoists fuel-loaded.
   4. Service platform hoist fuel-loaded.

APPLICABILITY: OPERATIONAL CONDITION 5* #.

ACTION:

a. With the reactor mode switch not locked in the Shutdown or Refuel position as specified, suspend CORE ALTERATIONS and lock the reactor mode switch in the Shutdown or Refuel position.

b. With the one-rod-out interlock inoperable, lock the reactor mode switch in the Shutdown position.

c. With any of the above required Refuel position equipment interlocks inoperable, suspend CORE ALTERATIONS with equipment associated with the inoperable Refuel position equipment interlock.

---

* See Special Test Exceptions 3.10.1 and 3.10.3.

# The reactor shall be maintained in OPERATIONAL CONDITION 5 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

333

SURVEILLANCE REQUIREMENTS

4.9.1.1 The reactor mode switch shall be verified to be locked in the Shutdown or Refuel position as specified:

a. Within 2 hours prior to:

1. Beginning CORE ALTERATIONS, and
2. Resuming CORE ALTERATIONS when the reactor mode switch has been unlocked.

b. At least once per 12 hours.

4.9.1.2 Each of the above required reactor mode switch Refuel position interlocks* shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST within 24 hours prior to the start of and at least once per 7 days during control rod withdrawal or CORE ALTERATIONS, as applicable.

4.9.1.3 Each of the above required reactor mode switch Refuel position interlocks* that is affected shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST prior to resuming control rod withdrawal or CORE ALTERATIONS, as applicable, following repair, maintenance or replacement of any component that could affect the Refuel position interlock.

---

*The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided that all control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 At least 2 source range monitor[*] (SRM) channels shall be OPERABLE and inserted to the normal operating level with:[##]

a. Annunciation and continuous visual indication in the control room,

b. One of the required SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other required SRM detector located in an adjacent quadrant, and

c. Unless adequate shutdown margin has been demonstrated per Specification 3.1.1, the "shorting links" removed from the RPS circuitry prior to and during the time any control rod is withdrawn.[#]

APPLICABILITY: OPERATIONAL CONDITION 5.[**]

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS and insert all insertable control rods.

SURVEILLANCE REQUIREMENTS

4.9.2 Each of the above required SRM channels shall be demonstrated OPERABLE by:

a. At least once per 12 hours:

  1. Performance of a CHANNEL CHECK,
  2. Verifying the detectors are inserted to the normal operating level, and
  3. During CORE ALTERATIONS, verifying that the detector of an OPERABLE SRM channel is located in the core quadrant where CORE ALTERATIONS are being performed and another is located in an adjacent quadrant.

---

[*] The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

[#] Not required for control rods removed per Specification 3.9.10.1 and 3.9.10.2.

[**] See Special Test Exception 3.10.7.

[##] Three SRM channels shall be OPERABLE for critical shutdown margin demonstrations. An SRM detector may be retracted provided a channel indication of at least 100 cps is maintained.

SURVEILLANCE REQUIREMENTS (Continued)

b. Performance of a CHANNEL FUNCTIONAL TEST:

   1. Within 24 hours prior to the start of CORE ALTERATIONS, and

   2. At least once per 7 days.

c. Verifying that the channel count rate is at least 0.7 cps:*

   1. Prior to control rod withdrawal,

   2. Prior to and at least once per 12 hours during CORE ALTERATIONS, and

   3. At least once per 24 hours.

d. Unless adequate shutdown margin has been demonstrated per Specification 3.1.1, verifying that the RPS circuitry "shorting links" have been removed, within 8 hours prior to and at least once per 12 hours during the time any control rod is withdrawn.**

---

*Provided signal-to-noise is $\geq$ 2. Otherwise, 3 cps.

**Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

3/4.9.3 CONTROL ROD POSITION

LIMITING CONDITION FOR OPERATION

3.9.3 All control rods shall be inserted.*

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.**

ACTION:

With all control rods not inserted, suspend all other CORE ALTERATIONS, except that one control rod may be withdrawn under control of the reactor mode switch Refuel position one-rod-out interlock.

SURVEILLANCE REQUIREMENTS

4.9.3 All control rods shall be verified to be inserted, except as above specified:

- a. Within 2 hours prior to:
  1. The start of CORE ALTERATIONS.
  2. The withdrawal of one control rod under the control of the reactor mode switch Refuel position one-rod-out interlock.
- b. At least once per 12 hours.

---

* Except control rods removed per Specification 3.9.10.1 or 3.9.10.2.
**See Special Test Exception 3.10.3.

337

3/4.9.4 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.4 The reactor shall be subcritical for at least 24 hours.

APPLICABILITY: OPERATIONAL CONDITION 5, during movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 24 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.4 The reactor shall be determined to have been subcritical for at least 24 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communication shall be maintained between the control room and refueling floor personnel.

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.

ACTION:

When direct communication between the control room and refueling floor personnel cannot be maintained, immediately suspend CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communication between the control room and refueling floor personnel shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

339

3/4.9.6 REFUELING PLATFORM

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling platform shall be OPERABLE and used for handling fuel assemblies or control rods within the reactor pressure vessel.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel.

ACTION:

With the requirements for refueling platform OPERABILITY not satisfied, suspend use of any inoperable refueling platform equipment from operations involving the handling of control rods and fuel assemblies within the reactor pressure vessel after placing the load in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.6.1 The refueling platform main hoist used for handling of control rods or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 7 days prior to the start of such operations by:

a. Demonstrating operation of the overload cutoff on the main hoist when the load exceeds 1200 + 0, -50 pounds.

b. Demonstrating operation of the main hoist uptravel stops when uptravel brings the top of active fuel to 8 feet below the normal water level.

c. Demonstrating operation of the slack cable cutoff on the main hoist when the load is less than 50 ± 10 pounds.

d. Demonstrating operation of the loaded rod block interlock on the main hoist when the load exceeds 485 ± 50 pounds.

e. Demonstrating operation of the redundant loaded interlock on the main hoist when the load exceeds 550 ± 50 pounds.

SURVEILLANCE REQUIREMENTS (Continued)

4.9.6.2 The refueling platform frame-mounted auxiliary hoist used for handling of control rods within the reactor pressure vessel shall be demonstrated OPERABLE within 7 days prior to the use of such equipment by:

a. Demonstrating operation of the overload cutoff on the frame mounted hoist when the load exceeds 500 ± 50 pounds.

b. Demonstrating operation of the uptravel mechanical stop on the frame mounted hoist when uptravel brings the grapple to 8 feet below the normal water level.

c. Demonstrating operation of the control rod block interlock on the frame mounted hoist when the load exceeds 400 ± 50 pounds.

4.9.6.3 The refueling platform monorail mounted auxiliary hoist used for handling of control rods within the reactor pressure vessel shall be demonstrated OPERABLE within 7 days prior to the use of such equipment by:

a. Demonstrating operation of the overload cutoff on the monorail hoist when the load exceeds 500 ± 50 pounds.

b. Demonstrating operation of the uptravel mechanical stop on the monorail hoist when uptravel brings the grapple to 8 feet below the normal water level.

c. Demonstrating operation of the control rod block interlock on the monorail hoist when the load exceeds 400 ± 50 pounds.

341

3/4.9.7 CRANE TRAVEL-SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 1200 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage pool racks unless handled by a single failure proof handling system.

APPLICABILITY: With fuel assemblies in the spent fuel storage pool racks.

ACTION:

With the requirements of the above specification not satisfied, place the polar crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7.1 Interlocks and physical stops which prevent polar crane main hoist travel over fuel assemblies in the spent fuel storage pool racks shall be demonstrated OPERABLE within 7 days prior to and at least once per 7 days during polar crane operation.

4.9.7.2 The single failure proof handling system shall be visually inspected and verified OPERABLE within 7 days prior to and at least once per 7 days during polar crane operation.

3/4 9.8 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.8 At least 22 feet 2 inches of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel while in OPERATIONAL CONDITION 5 when the fuel assemblies being handled are irradiated or the fuel assemblies seated within the reactor vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving handling of fuel assemblies or control rods within the reactor pressure vessel after placing all fuel assemblies and control rods in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.8 The reactor vessel water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours during handling of fuel assemblies or control rods within the reactor pressure vessel.

3/4.9.9 WATER LEVEL - SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.9 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area after placing the fuel assemblies and crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The water level in the spent fuel storage pool shall be determined to be at least at its minimum required depth at least once per 7 days.

3/4.9.10 CONTROL ROD REMOVAL

SINGLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.10.1 One control rod and/or the associated control rod drive mechanism may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until a control rod and associated control rod drive mechanism are reinstalled and the control rod is fully inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Table 1.2 and Specification 3.9.1.

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied, except that the control rod selected to be removed;

    1. May be assumed to be the highest worth control rod required to be assumed to be fully withdrawn by the SHUTDOWN MARGIN test, and

    2. Need not be assumed to be immovable or untrippable.

d. All other control rods in a five-by-five array centered on the control rod being removed are inserted and electrically or hydraulically disarmed or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

e. All other control rods are inserted.

f. All fuel loading operations shall be suspended.

APPLICABILITY: OPERATIONAL CONDITIONS 4 and 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of the control rod and/or associated control rod drive mechanism from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

SURVEILLANCE REQUIREMENTS

4.9.10.1 Within 4 hours prior to the start of removal of a control rod and/or the associated control rod drive mechanism from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until a control rod and associated control rod drive mechanism are reinstalled and the control rod is inserted in the core, verify that:

a. The reactor mode switch is OPERABLE per Surveillance Requirement 4.3.1.1 or 4.9.1.2, as applicable, and locked in the Shutdown position or in the Refuel position with the "one rod out" Refuel position interlock OPERABLE per Specification 3.9.1.

b. The SRM channels are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied per Specification 3.9.10.1.c.

d. All other control rods in a five-by-five array centered on the control rod being removed are inserted and electrically or hydraulically disarmed or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

e. All other control rods are inserted.

f. All fuel loading operations are suspended.

BEST COPY [illegible]

MULTIPLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.10.2 Any number of control rods and/or control rod drive mechanisms may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Specification 3.9.1, except that the Refuel position "one-rod-out" interlock may be bypassed, as required, for those control rods and/or control rod drive mechanisms to be removed, after the fuel assemblies have been removed as specified below.

b. The source range monitors SRM are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

f. All fuel loading operations shall be suspended.

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

347

SURVEILLANCE REQUIREMENTS

4.9.10.2.1 Within 4 hours prior to the start of removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core, verify that:

a. The reactor mode switch is OPERABLE per Surveillance Requirement 4.3.1.1 or 4.9.1.2, as applicable, and locked in the Shutdown position or in the Refuel position per Specification 3.9.1.

b. The SRM channels are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod and/or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

f. All fuel loading operations are suspended.

4.9.10 2.2 Following replacement of all control rods and/or control rod drive mechanisms removed in accordance with this specification, perform a functional test of the "one-rod-out" Refuel position interlock, if this function had been bypassed.

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11.1 At least one shutdown cooling mode loop of the residual heat removal (RHR) system shall be OPERABLE and in operation* with:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is greater than or equal to 22 feet 2 inches above the top of the reactor pressure vessel flange and heat losses to ambient** are not sufficient to maintain OPERATIONAL CONDITION 5.

ACTION:

a. With no RHR shutdown cooling mode loop OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal. Otherwise, suspend all operations involving an increase in the reactor decay heat load and establish SECONDARY CONTAINMENT INTEGRITY within 4 hours.

b. With no RHR shutdown cooling mode loop in operation, within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature at least once per hour.

SURVEILLANCE REQUIREMENTS

4.9.11.1 At least one shutdown cooling mode loop of the residual heat removal system or alternate method shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

---

*The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

**Ambient losses must be such that no increase in reactor vessel water temperature will occur (even though REFUELING conditions are being maintained).

349

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11.2 Two shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and at least one loop shall be in operation,* with each loop consisting of:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is less than 22 feet 2 inches above the top of the reactor pressure vessel flange and heat losses to ambient** are not sufficient to maintain OPERATIONAL CONDITION 5.

ACTION:

a. With less than the above required shutdown cooling mode loops of the RHR system OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the OPERABILITY of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop.

b. With no RHR shutdown cooling mode loop in operation, within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature at least once per hour.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11.2 At least one shutdown cooling mode loop of the residual heat removal system or alternate method shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

---

*The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

**Ambient losses must be such that no increase in reactor vessel water temperature will occur (even though REFUELING conditions are being maintained).

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.10.1 The provisions of Specifications 3.6.1.1, 3.6.1.3 and 3.9.1 and Table 1.2 may be suspended to permit the reactor pressure vessel closure head and the drywell head to be removed and the primary containment air lock doors to be open when the reactor mode switch is in the Startup position during low power PHYSICS TESTS with THERMAL POWER less than 1% of RATED THERMAL POWER and reactor coolant temperature less than 200°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during low power PHYSICS TESTS.

ACTION:

With THERMAL POWER greater than or equal to 1% of RATED THERMAL POWER or with the reactor coolant temperature greater than or equal to 200°F, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.1 The THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour during low power PHYSICS TESTS.

3/4.10.2 ROD SEQUENCE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.10.2 The sequence constraints imposed on control rod groups by the rod worth minimizer (RWM) per Specification 3.1.4.1 and by the rod sequence control system (RSCS) per Specification 3.1.4.2 may be suspended by means of bypass switches for the following tests provided that control rod movement prescribed for this testing is verified by a second licensed operator or other technically qualified member of the unit technical staff present at the reactor console:

a. Shutdown margin demonstrations, Specification 4.1.1.

b. Control rod scram, Specification 4.1.3.2.

c. Control rod friction measurements.

d. Startup Test Program with the THERMAL POWER less than 20% of RATED THERMAL POWER.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the requirements of the above specification not satisfied, verify that the RWM and/or the RSCS is OPERABLE per Specifications 3.1.4.1 and 3.1.4.2, respectively.

SURVEILLANCE REQUIREMENTS

4.10.2 When the sequence constraints imposed by the RSCS and/or RWM are bypassed, verify:

a. That movement of the control rods from 75% ROD DENSITY to the RSCS low power setpoint is limited to the approved control rod withdrawal sequence during scram and friction tests.

b. That movement of control rods during shutdown margin demonstrations is limited to the prescribed sequence per Specification 3.10.3.

c. Conformance with this specification and test procedures by a second licensed operator or other technically qualified member of the unit technical staff.

BEST COPY

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

LIMITING CONDITION FOR OPERATION

3.10.3 The provisions of Specification 3.9.1, Specification 3.9.3 and Table 1.2 may be suspended to permit the reactor mode switch to be in the Startup position and to allow more than one control rod to be withdrawn for shutdown margin demonstration, provided that at least the following requirements are satisfied.

a. The source range monitors are OPERABLE with the RPS circuitry "shorting links" removed per Specification 3.9.2.

b. The rod worth minimizer is OPERABLE per Specification 3.1.4.1 and is programmed for the shutdown margin demonstration, or conformance with the shutdown margin demonstration procedure is verified by a second licensed operator or other technically qualified member of the unit technical staff.

c. The "rod-out-notch-override" control shall not be used during out-of-sequence movement of the control rods.

d. No other CORE ALTERATIONS are in progress.

APPLICABILITY: OPERATIONAL CONDITION 5, during shutdown margin demonstrations.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown or Refuel position.

SURVEILLANCE REQUIREMENTS

4.10.3 Within 30 minutes prior to and at least once per 12 hours during the performance of a shutdown margin demonstration, verify that;

a. The source range monitors are OPERABLE per Specification 3.9.2,

b. The rod worth minimizer is OPERABLE with the required program per Specification 3.1.4.1 or a second licensed operator or other technically qualified member of the unit technical staff is present and verifies compliance with the shutdown demonstration procedures, and

c. No other CORE ALTERATIONS are in progress.

SPECIAL TEST EXCEPTIONS

3/4.10.4 RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.4.1.1 and 3.4.1.3 that recirculation loops be in operation with matched pump speed may be suspended for up to 24 hours for the performance of:

a. PHYSICS TESTS, provided that THERMAL POWER does not exceed 5% of RATED THERMAL POWER, or

b. The Startup Test Program.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2, during PHYSICS TESTS and the Startup Test Program.

ACTION:

a. With the above specified time limit exceeded, insert all control rods.

b. With the above specified THERMAL POWER limit exceeded during PHYSICS TESTS, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The time during which the above specified requirement has been suspended shall be verified to be less than 24 hours at least once per hour during PHYSICS TESTS and the Startup Test Program.

4.10.4.2 THERMAL POWER shall be determined to be less than 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

3/4.10.5 OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.10.5 The provisions of Specification 3.6.6.2 may be suspended during the performance of the Startup Test Program until 6 months after initial criticality.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION

With the requirements of the above specification not satisfied, be in at least STARTUP within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.5 The number of months since initial criticality shall be verified to be less than or equal to 6 months at least once per 31 days during the Startup Test Program.

355

SPECIAL TEST EXCEPTIONS

3/4.10.6 TRAINING STARTUPS

LIMITING CONDITION FOR OPERATION

3.10.6 The provisions of Specification 3.5.1 may be suspended to permit one RHR subsystem to be aligned in the shutdown cooling mode during training startups provided that the reactor vessel is not pressurized, THERMAL POWER is less than or equal to 1% of RATED THERMAL POWER and reactor coolant temperature is less than 200°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during training startups.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.6 The reactor vessel shall be verified to be unpressurized and the THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour during training startups.

3/4.10.7 SPECIAL INSTRUMENTATION - INITIAL CORE LOADING

LIMITING CONDITION FOR OPERATION

3.10.7 During initial core loading within the Startup Test Program the provisions of Specification 3.9.2 may be suspended provided that at least two source range monitor (SRM) channels with detectors inserted to the normal operating level are OPERABLE with:

a. One of the required SRM channels continuously indicating* in the control room,

b. One of the required SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other required SRM detector located in an adjacent quadrant,**

c. The RPS "shorting links" shall be removed prior to and during fuel loading,

d. The reactor mode switch is OPERABLE and locked in the REFUEL position.

APPLICABILITY: OPERATIONAL CONDITION 5

ACTION

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS and insert all insertable control rods.

SURVEILLANCE REQUIREMENTS

4.10.7 Each of the above required SRM channels shall be demonstrated OPERABLE by:

a. Within 1 hour prior to and at least once per 12 hours during CORE ALTERATIONS:

   1. Performance of a CHANNEL CHECK***

   2. Confirming that the above required SRM detectors are at the normal operating level and located in the quadrants required by Specification 3.10.7.

---

*Up to 16 fuel bundles may be loaded without a visual indication of count rate.

**The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

***Check may be performed by use of movable neutron source.

4.10.7. (Continued)

3. The RPS "shorting links" are removed.

4. The reactor mode switch is locked in the REFUEL position.

b. Performance of a CHANNEL FUNCTIONAL TEST within 24 hours prior to the start and at least once per 7 days during CORE ALTERATIONS.

c. Verifying for at least one SRM channel that the count rate is at least 0.7 cps*:

1. Immediately following the loading of the first 16 fuel bundles.

2. At least once per 12 hours thereafter during CORE ALTERATIONS.

---

*Provided signal-to-noise is ≥ 2. Otherwise, 3 cps.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 5.1.1-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microcuries/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11.1.1.1-1.

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

TABLE 4.11.1.1.1-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (μCi/ml)
A. Batch Waste Release[b] Sample Tanks (3)	P Each Batch	P Each Batch	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	P Each Batch	M Composite[d]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite[d]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
B. Continuous Releases[e] Station Service Water System (GSW) (If contaminated as indicated by SACS system)	NA	M Composite[d]	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
	W Crab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	NA	M Composite[d]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	NA	Q Composite[d]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

360

TABLE 4.11.1.1.1-1 (Continued)

TABLE NOTATION

[a] The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$\text{LLD} = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as microcuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide (sec^{-1}), and

Δt for plant effluents is the elapsed time between the midpoint of sample collection and time of counting (sec).

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

[b] A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed by a method described in the ODCM to assure representative sampling.

361

TABLE 4.11.1.1.1-1 (Continued)

TABLE NOTATION

[c] The principal gamma emitters for which the LLD specification applies exclusively are: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, and Ce-141. Ce-144 shall also be measured, but with an LLD of 5×10^{-6}. This does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.7.

[d] A composite sample is one in which the quantity of liquid samples is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released.

[e] A continuous release is the discharge of liquid wastes of a nondiscrete volume; e.g., from a volume of a system that has an input flow during the continuous release.

DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each reactor unit, to UNRESTRICTED AREAS (see Figure 5.1.1-1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the total body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the total body and to less than or equal to 10 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.2 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

LIMITING CONDITION FOR OPERATION

3.11.1.3 The liquid radwaste treatment system shall be OPERABLE and appropriate portions of the system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses due to the liquid effluent, from each reactor unit, to UNRESTRICTED AREAS (see Figure 5.1.1-1) would exceed 0.06 mrem to the total body or 0.2 mrem to any organ in any 31-day period.

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid waste being discharged and in excess of the above limits and any portion of the liquid radwaste treatment system not in operation, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report that includes the following information:

    1. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

    2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

    3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases from each reactor unit to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM.

4.11.1.3.2 The installed liquid radwaste treatment system shall be demonstrated OPERABLE by meeting Specifications 3.11.1.1 and 3.11.1.2.

Jle 4

RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in any outside temporary tank shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.7.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1.1-1) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the total body and less than or equal to 3000 mrems/yr to the skin, and

b. For iodine-131, iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately restore the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM.

4.11.2.1.2 The dose rate due to iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11.2.1.2-1.

366

TABLE 4.11.2.1.2-1

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (µCi/ml)
A. Containment PURGE	P Each PURGE(c)	P Each PURGE(c)	Principal Gamma Emitters(b)	1×10^{-4}
	Grab Sample	P	H-3 (oxide)	1×10^{-6}
B. North Plant Vent South Plant Vent	M(c),(d) Grab Sample	M(c)	Principal Gamma Emitters(b) H-3 (oxide)	1×10^{-4} 1×10^{-6}
C. All Release Types as listed in A and B above.	Continuous(e)	W(f) Charcoal Sample	I-131	1×10^{-12}
	Continuous(e)	W(f) Particulate Sample	Principal Gamma Emitters(b)	1×10^{-11}
	Continuous(e)	Q Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous(e)	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous(e)	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}

TABLE 4.11.2.1.2-1 (Continued)

TABLE NOTATION

(a)The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$\text{LLD} = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as microcuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide (sec^{-1}), and

Δt for plant effluents is the elapsed time between the midpoint of sample collection and time of counting (sec).

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

BEST COPY

TABLE 4.11.2.1.2-1 (Continued)

TABLE NOTATIONS

(b)The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 in noble gas releases and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, I-131, Cs-134, Cs-137, Ce-141 and Ce-144 in iodine and particulate releases. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.7.

(c)Sampling and analysis shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15% of RATED THERMAL POWER within a 1-hour period. This requirement does not apply if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

(d)Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

(e)The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2, and 3.11.2.3.

(f)Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing, or after removal from sampler. Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15% of RATED THERMAL POWER in 1 hour and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

369

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each reactor unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1.1-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation and,

b. During any calendar year: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.

APPLICABILITY: At all times.

ACTION

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.2 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each reactor unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1.1-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

b. During any calendar year: Less than or equal to 15 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of iodine-131, iodine-133, tritium, and radionuclides in particulate form with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.3 Cumulative dose contributions for the current calendar quarter and current calendar year for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

371

GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE TREATMENT SYSTEM shall be in operation.

APPLICABILITY: Whenever the main condenser steam jet air ejector system is in operation.

ACTION:

a. With gaseous radwaste from the main condenser air ejector system being discharged without treatment for more than 7 days, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for the inoperability,
   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and
   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.4 The readings of the relevant instruments shall be checked every 12 hours when the main condenser air ejector is in use to ensure that the gaseous radwaste treatment system is functioning.

312

VENTILATION EXHAUST TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.2.5 The VENTILATION EXHAUST TREATMENT SYSTEM for the Reactor Building and the Service and Radwaste Building shall be OPERABLE and appropriate portions of this system shall be used to reduce release of radioactivity when the projected doses in 31 days due to gaseous effluent releases from each unit to areas at and beyond the SITE BOUNDARY (see Figure 5.1.1-1) would exceed:

a. 0.2 mrad to air from gamma radiation, or

b. 0.4 mrad to air from beta radiation, or

c. 0.3 mrem to any organ of a MEMBER OF THE PUBLIC

APPLICABILITY: At all times.

ACTION:

a. With radioactive ventilation exhaust being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report that includes the following information:

   1. Identification of any inoperable equipment or subsystems, and the reason for the inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5.1 Doses due to gaseous releases from each unit to areas at and beyond the SITE BOUNDARY shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM, when the VENTILATION EXHAUST TREATMENT SYSTEM is not being fully utilized.

4.11.2.5.2 The installed VENTILATION EXHAUST TREATMENT SYSTEM shall be considered OPERABLE by meeting Specifications 3.11.2.1 and 3.11.2.2 and 3.11.2.3.

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.6 The concentration of hydrogen in the main condenser offgas treatment system shall be limited to less than or equal to 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of hydrogen in the main condenser offgas treatment system exceeding the limit, restore the concentration to within the limit within 48 hours.

b. With continuous monitors inoperable, operation of the main condenser offgas treatment system may continue for up to 30 days provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The concentration of hydrogen in the main condenser offgas treatment system shall be determined to be within the above limits by continuously monitoring the waste gases in the main condenser offgas treatment system whenever the main condenser evacuation system is in operation with the hydrogen monitors required OPERABLE by Table 3.3.7.10-1 of Specification 3.3.7.10.

RADIOACTIVE EFFLUENTS

MAIN CONDENSER

LIMITING CONDITION FOR OPERATION

3.11.2.7 The radioactivity rate of noble gases measured at the recombiner after-condenser discharge shall be limited to less than or equal to 330 millicuries/sec after 30 minute decay.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3*.

ACTION:

With the radioactivity rate of noble gases at the recombiner after-condenser discharge exceeding 330 millicuries/sec after 30 minute decay, restore the radioactivity rate to within its limit within 72 hours or be in at least HOT STANDBY within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.11.2.7.1 The radioactivity rate of noble gases at the recombiner after-condenser discharge shall be continuously monitored in accordance with Specification 3.3.7.1.

4.11.2.7.2 The radioactivity rate of noble gases from the recombiner after-condenser discharge shall be determined to be within the limits of Specification 3.11.2.7 at the following frequencies by performing an isotopic analysis of a representative sample of gases taken near the discharge of the main condenser air ejector:

a. At least once per 31 days.

b. Within 4 hours following an increase, as indicated by the Offgas Radioactivity Monitor, of greater than 50%, after factoring out increases due to changes in THERMAL POWER level, in the nominal steady-state fission gas release from the primary coolant.

c. The provisions of Specification 4.0.4 are not applicable.

---

*When the main condenser air ejector is in operation.

VENTING OR PURGING

LIMITING CONDITION FOR OPERATION

3.11.2.8 VENTING or PURGING of the Mark I containment drywell shall be through either the reactor building ventilation system or the filtration, recirculation and ventilation system.

APPLICABILITY: Whenever the containment is vented or purged.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all VENTING and PURGING of the drywell.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.8 The containment shall be determined to be aligned for VENTING or PURGING through either the reactor building ventilation system or the filtration, recirculation and ventilation system within 4 hours prior to start of and at least once per 12 hours during VENTING or PURGING of the drywell.

3/4.11.3 SOLID RADIOACTIVE WASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.3 Radioactive wastes shall be SOLIDIFIED or dewatered in accordance with the PROCESS CONTROL PROGRAM to meet shipping and transportation requirements during transit, and disposal site requirements when received at the disposal site.

APPLICABILITY: At all times.

ACTION:

a. With SOLIDIFICATION or dewatering not meeting disposal site and shipping and transportation requirements, suspend shipment of the inadequately processed wastes and correct the PROCESS CONTROL PROGRAM, the procedures and/or the solid waste system as necessary to prevent recurrence.

b. With SOLIDIFICATION or dewatering not performed in accordance with the PROCESS CONTROL PROGRAM, (1) demonstrate by test or analysis that the improperly processed waste in each container meets the requirements for transportation to the disposal site and for receipt at the disposal site and (2) take appropriate administrative action to prevent recurrence.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.3.1 The PROCESS CONTROL PROGRAM shall be followed to verify that the properties of the packaged waste meet the minimum stability requirements of 10 CFR Part 61 and other requirements for transportation to the disposal site and receipt at the disposal site.

4.11.3.2 The PROCESS CONTROL PROGRAM shall include sufficient quality control and assurance methods to assure the solidified waste product from any process (either in-house or contracted vendor) meets the requirements for transportation and receipt at the disposal site.

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2a., 3.11.1.2b., 3.11.2.2a., 3.11.2.2b., 3.11.2.3a., or 3.11.2.3b., calculations should be made including direct radiation contributions from the units including outside storage tanks, etc. to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from the units including outside storage tanks, etc. shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in Specification 3.11.4, ACTION a.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12.1-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12.1-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.7, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12.1-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose* to A MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, and 3.11.2.3. When more than one of the radionuclides in Table 3.12.1-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots > 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* to A MEMBER OF THE PUBLIC from all radionuclides is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 3.12.1-1, identify specific locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days.

---

*The methodology used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

The specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.9.1.8, identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8 and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12.1-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12.1-1 and the detection capabilities required by Table 4.12.1-1.

TABLE 3.12.1-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM*

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations(1)	Sampling and Collection Frequency	Type and Frequency of Analysis
1. DIRECT RADIATION(2)	Forty-three routine monitoring stations either with two or more dosimeters placed as follows: An inner ring of stations, one in each meteorological sector in the general area of the SITE BOUNDARY; An outer ring of stations, one in each land based meteorological sector in the 6- to 8-km range from the site; and The balance of the stations to be placed in special interest areas such as population centers, nearby residences, schools, and in one or two areas to serve as control stations.	Quarterly	Gamma dose quarterly.

---

*The number, media, frequency, and location of samples may vary from site to site. This table presents an acceptable minimum program for a site at which each entry is applicable. Local site characteristics must be examined to determine if pathways not covered by this table may significantly contribute to an individual's dose and should be included in the sample program.

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations(1)	Sampling and Collection Frequency	Type and Frequency of Analysis
2. AIRBORNE			
Radioiodine and Particulates	Samples from 5 locations. Three samples from close to the SITE BOUNDARY locations, in different sectors, of a high calculated annual average ground-level D/Q. One sample from the vicinity of a community having a high calculated annual average groundlevel D/Q. One sample from a control location, as for example 15-30 km distant and in the least prevalent wind direction.	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Cannister: I-131 analysis weekly. Particulate Sampler: Gross beta radioactivity analysis following filter change;(3) Gamma isotopic analysis(4) of composite (by location) quarterly.
3. WATERBORNE			
a. Surface(5)	One sample upstream. One sample downstream. One sample crosstream.	Grab sample monthly.	Gamma isotopic analysis(4) monthly. Composite for tritium analysis quarterly.
b. Ground	Samples from one or two sources only if likely to be affected(7).	Monthly	Gamma isotopic(4) and tritium analysis monthly.

BLANK PAGE

BLANK PAGE

BLANK PAGE

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations(1)	Sampling and Collection Frequency	Type and Frequency of Analysis
c. Drinking	One sample of each of one to three the nearest water supplies that could be affected by its discharge. One sample from a control location.	Composite sample over 2-week period(6) when I-131 analysis is performed, monthly composite otherwise	I-131 analysis on each composite when the dose calculated for the consumption of the water is greater than 1 mrem per year.(8) Composite for gross beta and gamma isotopic analyses(4) monthly. Composite for tritium analysis quarterly.
d. Sediment	One sample from downstream area One sample from upstream area One sample from cross stream area	Semiannually	Gamma isotopic analysis(4) semiannually.
4. INGESTION			
a. Milk	Samples from milking animals in three locations within 5 km distance having the highest dose potential. If there are none, then, 1 sample from milking animals in each of three areas between 5 to 8 km distant where doses are calculated to be greater than 1 mrem per yr.(8) One sample from milking animals at a control location 15-30 km distant.	Semimonthly when animals are on pasture, monthly at other times	Gamma isotopic monthly(4) and I-131 analysis semimonthly when animals are on pasture; monthly at other times.

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations(1)	Sampling and Collection Frequency	Type and Frequency of Analysis
4. INGESTION (Continued)			
b. Fish and Invertebrates	One sample of each commercially and recreationally important species in vicinity of plant discharge area. One sample of same species in areas not influenced by plant discharge.	Sample in season, or semiannually if they are not seasonal	Gamma isotopic analysis(4) on edible portions.
c. Food Products	One sample of each principal class of food products from any area that is irrigated by water in which liquid plant wastes have been discharged.	At time of harvest(9)	Gamma isotopic analysis(4) on edible portion.

384

TABLE 3.12.1-1 (Continued)

TABLE NOTATIONS

(1)Specific parameters of distance and direction sector from the centerline of one reactor, and additional description where pertinent, shall be provided for each and every sample location in Table 3.12.1-1 in a table and figure(s) in the ODCM. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978, and to Radiological Assessment Branch Technical Position, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable specific alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the Radiological Environmental Monitoring Program given in the ODCM. Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table for the ODCM reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples for that pathway and justifying the selection of the new location(s) for obtaining samples.

(2)One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading.

(3)Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than 10 times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

(4)Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

(5)The "upstream sample" shall be taken at a distance beyond significant influence of the discharge. The "downstream" sample shall be taken in an area beyond but near the mixing zone. "Upstream" samples in an estuary must be taken far enough upstream to be beyond the plant influence. Salt water shall be sampled only when the receiving water is utilized for recreational activities.

TABLE 3.12.1-1 (Continued)

TABLE NOTATION

(6)A composite sample is one in which the quantity (aliquot) of liquid sampled is proportional to the quantity of flowing liquid and in which the method of sampling employed results in a specimen that is representative of the liquid flow. In this program composite sample aliquots shall be collected at time intervals that are very short relative to the compositing period in order to assure obtaining a representative sample.

(7)Groundwater samples shall be taken when this source is tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination.

(8)The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

(9)If harvest occurs more than once a year, sampling shall be performed during each discrete harvest. If harvest occurs continuously, sampling shall be monthly. Attention shall be paid to including samples of tuberous and root food products.

TABLE 3.12.1-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

REPORTING LEVELS

Analysis	Water (pCi/ℓ)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/ℓ)	Food Products (pCi/kg, wet)
H-3	30,000				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95	400				
I-131	2	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140	200			300	

TABLE 4.12.1-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS(1)(2)

LOWER LIMIT OF DETECTION (LLD)(3)

Analysis	Water (pCi/ℓ)	Airborne Particulate or Gas (pCi/m^3)	Fish (pCi/kg,wet)	Milk (pCi/ℓ)	Food Products (pCi/kg,wet)	Sediment (pCi/kg,dry)
gross beta	4	0.01				
H-3	3000					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zn-65	30		260			
Zr-Nb-95	15					
I-131	1	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-La-140	15			15		

TABLE 4.12.1-1 (Continued)

TABLE NOTATIONS

(1)This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

(2)Required detection capabilities for thermoluminescent dosimeters used for environmental measurements shall be in accordance with the recommendations of Regulatory Guide 4.13.

(3)The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$\text{LLD} = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as picocuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide (sec^{-1}), and

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period, and time of counting (sec)

Typical valves of E, V, Y, and Δt should be used in the calculation.

389

TABLE 4.12.1-1 (Continued)

TABLE NOTATIONS

It should be recognized that the LLD is defined as an *a priori* (before the fact) limit representing the capability of a measurement system and not as an *a posteriori* (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence and the nearest garden* of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation.

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.7.

b. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) to the radiological environmental monitoring program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. Pursuant to Specification 6.9.1.8, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, visual survey, aerial survey, or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

---

*Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the SITE BOUNDARY in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 3.12.1-1, Part 4.c., shall be followed, including analysis of control samples.

AVAILABLE

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.3 The Interlaboratory Comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Inter laboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

BASES FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The Summary statements contained in this section provide the bases for the specifications in Section 3.0 and 4.0 but in accordance with 10 CFR 50.36 are not a part of these Technical Specifications.

3/4.0 APPLICABILITY

BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4.

3.0.1 This specification states the applicability of each specification in terms of defined OPERATIONAL CONDITION or other specified applicability condition and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 This specification delineates the measures to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of the specification. For example, Specification 3.7.2 calls for two control room emergency filtration subsystems to be OPERABLE and provides explicit ACTION requirements if one subsystem is inoperable. Under the requirements of Specification 3.0.3, if both of the required subsystems are inoperable, within one hour measures must be initiated to place the unit in at least STARTUP within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours and in COLD SHUTDOWN within the subsequent 24 hours. As a further example, Specification 3.6.6.1 requires two containment hydrogen recombiner systems to be OPERABLE and provides explicit ACTION requirements if one recombiner system is inoperable. Under the requirements of Specification 3.0.3, if both of the required systems are inoperable, within one hour measures must be initiated to place the unit in at least STARTUP within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

3.0.4 This specification provides that entry into an OPERATIONAL CONDITION must be made with (a) the full complement of required systems, equipment or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out of service provisions contained in the ACTION statements.

The intent of this provision is to ensure that unit operation is not initiated with either required equipment or systems inoperable or other limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

4.0.1 This specification provides that surveillance activities necessary to ensure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL CONDITIONS or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL CONDITIONS or other conditions are provided in the individual Surveillance Requirements. Surveillance Requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance; instead, it permits the more frequent performance of surveillance activities.

The tolerance values, taken either individually or consecutively over 3 test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under this criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE, when such items are found or known to be inoperable although still meeting the Surveillance Requirements.

4.0.4 This specification ensures that surveillance activities associated with a Limiting Conditions for Operation have been performed within the specified time interval prior to entry into an applicable OPERATIONAL CONDITION or other specified applicability condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outage, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50, Section 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies of performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL CONDITION or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

3/4.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed in the cold, xenon-free condition and shall show the core to be subcritical by at least R + 0.38% delta k/k or R + 0.28% delta k/k, as appropriate. The value of R in units of % delta k/k is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of R must be positive or zero and must be determined for each fuel loading cycle.

Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of demonstration of the SHUTDOWN MARGIN. The highest worth rod may be determined analytically or by test. The SHUTDOWN MARGIN is demonstrated by an insequence control rod withdrawal at the beginning of life fuel cycle conditions, and, if necessary, at any future time in the cycle if the first demonstration indicates that the required margin could be reduced as a function of exposure. Observation of subcriticality in this condition assures subcriticality with the most reactive control rod fully withdrawn.

This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion.

3/4.1.2 REACTIVITY ANOMALIES

Since the SHUTDOWN MARGIN requirement for the reactor is small, a careful check on actual conditions to the predicted conditions is necessary, and the changes in reactivity can be inferred from these comparisons of rod patterns. Since the comparisons are easily done, frequent checks are not an imposition on normal operations. A 1% delta k/k change is larger than is expected for normal operation so a change of this magnitude should be thoroughly evaluated. A change as large as 1% delta k/k would not exceed the design conditions of the reactor and is on the safe side of the postulated transients.

398

3/4.1.3 CONTROL RODS

The specifications of this section ensure that (1) the minimum SHUTDOWN MARGIN is maintained, (2) the control rod insertion times are consistent with those used in the accident analysis, and (3) limit the potential effects of the rod drop accident. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem, therefore with a withdrawn control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the nonfully-inserted position are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shutdown for investigation and resolution of the problem.

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than 1.06 during the limiting power transient analyzed in Section 15.4 of the FSAR. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than 1.06. The occurrence of scram times longer then those specified should be viewed as an indication of a systematic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required.

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. Operability of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.

CONTROL RODS (Continued)

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after completing CORE ALTERATIONS that could have affected the control rod coupling integrity. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

The control rod housing support restricts the outward movement of a control rod to less than 6 inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 20% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RSCS and/or RWM to be OPERABLE when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER provides adequate control.

The RSCS and RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section 15.4.9 of the FSAR and the techniques of the analysis are presented in a topical report, Reference 1, and two supplements, References 2 and 3.

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods.

400

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that the withdrawn control rods remain fixed in the rated power pattern. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core and other piping systems connected to the reactor vessel. To allow for potential leakage and imperfect mixing, this concentration is increased by 25%. The required concentration is achieved by having a minimum available quantity of 4850 gallons of sodium-pentaborate solution containing a minimum of 5760 lbs of sodium-pentaborate. This quantity of solution is a net amount which is above the pump suction, thus allowing for the portion which cannot be injected. The pumping rate of 41.2 gpm per pump provides a negative reactivity insertion rate over the permissible pentaborate solution volume range, which adequately compensates for the positive reactivity effects due to temperature and Xenon during shutdown. The temperature requirement is necessary to ensure that the sodium-pentaborate remains in solution.

With redundant pumps and explosive injection valves and with a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the redundant components inoperable.

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use.

Replacement of the explosive charges in the valves at regular intervals will assure that these valves will not fail because of deterioration of the charges.

---

1. C. J. Paone, R. C. Stirn and J. A. Woolley, "Rod Drop Accident Analysis for Large BWR's", G. E. Topical Report NEDO-10527, March 1972
2. C. J. Paone, R. C. Stirn and R. M. Young, Supplement 1 to NEDO-10527, July 1972
3. J. M. Haun, C. J. Paone and R. C. Stirn, Addendum 2, "Exposed Cores", Supplement 2 to NEDO-10527, January 1973

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant_accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4 and 3.2.1-5.

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4 and 3.2.1-5 is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis compared to previous analyses can be broken down as follows.

a. Input Changes

1. Corrected Vaporization Calculation - Coefficients in the vaporization correlation used in the REFLOOD code were corrected.

2. Incorporated more accurate bypass areas - The bypass areas in the top guide were recalculated using a more accurate technique.

3. Corrected guide tube thermal resistance.

4. Corrected heat capacity of reactor internals heat nodes.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

b. Model Change

1. Core CCFL pressure differential - 1 psi - Incorporate the assumption that flow from the bypass to lower plenum must overcome a 1 psi pressure drop in core.

2. Incoporate NRC pressure transfer assumption - The assumption used in the SAFE-REFLOOD pressure transfer when the pressure is increasing was changed.

A few of the changes affect the accident calculation irrespective of CCFL. These changes are listed below.

a. Input Change

1. Break Areas - The DBA break area was calculated more accurately.

b. Model Change

1. Improved Radiation and Conduction Calculation - Incorporation of CHASTE 05 for heatup calculation.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Bases Table B 3.2.1-1.

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow biased simulated thermal power-upscale scram setting and the flow biased neutron flux-upscale control rod block trip setpoints must be adjusted to ensure that the MCPR does not become less than 1.06 or that > 1% plastic strain does not occur in the degraded situation. The scram setpoints and rod block setpoints are adjusted in accordance with the formula in Specification 3.2.2 whenever it is known that the existing power distribution would cause the design LHGR to be exceeded at RATED THERMAL POWER.

BEST COPY A

Bases Table B 3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS

Plant Parameters:

Core THERMAL POWER	3430 Mwt* which corresponds to 105% of rated steam flow
Vessel Steam Output	14.87×10^6 lbm/hr which corresponds to 105% of rated steam flow
Vessel Steam Dome Pressure	1055 psia

Design Basis Recirculation Line
Break Area for:
a. Large Breaks 4.1 ft^2

b. Small Breaks 0.09 ft^2,

Fuel Parameters:

FUEL TYPE	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kw/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO
Initial Core	8 x 8	13.4	1.4	1.20

A more detailed listing of input of each model and its source is presented in Section II of Reference 1 and subsection 6.3.3 of the FSAR.

*This power level meets the Appendix K requirement of 102%. The core heatup calculation assumes a bundle power consistent with operation of the highest powered rod at 102% of its Technical Specification LINEAR HEAT GENERATION RATE limit.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR of 1.06, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR of 1.06, the required minimum operating limit MCPR of Specification 3.2.3 is obtained.

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.0-3 that are input to a GE-core dynamic behavior transient computer program. The code used to evaluate pressurization events is described in NEDO-24154(3) and the program used in non-pressurization events is described in NEDO-10802(2). The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic TASC code described in NEDE-25149(4). The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the K_f factor of Figure 3.2.3-2 is to define operating limits at other than rated core flow conditions. At less than 100% of rated flow the required MCPR is the product of the MCPR and the K_f factor. The K_f factors assure that the Safety Limit MCPR will not be violated during a flow increase transient resulting from a motor-generator speed control failure. The K_f factors may be applied to both manual and automatic flow control modes.

The K_f factors values shown in Figure 3.2.3-2 were developed generically and are applicable to all BWR/2, BWR/3 and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to RATED THERMAL POWER at rated core flow.

For the manual flow control mode, the K_f factors were calculated such that for the maximum flow rate, as limited by the pump scoop tube set point and the corresponding THERMAL POWER along the rated flow control line, the limiting bundle's relative power was adjusted until the MCPR changes with different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR, determines the K_f.

405

MINIMUM CRITICAL POWER RATIO (Continued)

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at RATED THERMAL POWER and rated thermal flow.

The K_f factors shown in Figure 3.2.3-2 are conservative for the General Electric plant operation because the operating limit MCPRs of Specification 3.2.3 is the same as the original 1.20 operating limit MCPR used for the generic derivation of K_f.

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the Linear Heat Generation Rate (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated.

References:

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566, November 1975.
2. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, NEDO-10802, February 1973.
3. Qualification of the One Dimensional Core Transient Model for Boiling Water Reactors, NEDO-24154, October 1978.
4. TASC 01-A Computer Program for the Transient Analysis of a Single Channel, Technical Description, NEDE-25149, January 1980.

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279 for nuclear power plant protection systems. The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the safety analyses. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

407

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance. Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For D.C. operated valves, a 3 second delay is assumed before the valve starts to move. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 13 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C. operated valve is assumed; thus the signal delay (sensor response) is concurrent with the 10 second diesel startup. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 13 second delay. It follows that checking the valve speeds and the 13 second time for emergency power establishment will establish the response time for the isolation functions.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971, NEDO-24222, dated December 1979, and Section 15.8 of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 30% of RATED THERMAL POWER are annunciated in the control room.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 175 ms. Included in this time are: the response time of the sensor, the time allotted for breaker arc suppression (135 ms @ 100% RTP), and the response time of the system logic.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits and Section 3/4.3 Instrumentation. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels, and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with 10 CFR Part 50, Appendix A, General Design Criteria 19, 41, 60, 61, 63 and 64.

3.4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

The OPERABILITY of the seismic monitoring instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit. This instrumentation is consistent with the recommendations of Regulatory Guide 1.12 "Instrumentation for Earthquakes," April 1974

3/4.3.7.3 METEOROLOGICAL MONITORING INSTRUMENTATION

The OPERABILITY of the meteorological monitoring instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of

MONITORING INSTRUMENTATION (Continued)

radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February, 1972.

3/4.3.7.4 REMOTE SHUTDOWN MONITORING INSTRUMENTATION AND CONTROLS

The OPERABILITY of the remote shutdown monitoring instrumentation and controls ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the unit from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1980 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.7.6 SOURCE RANGE MONITORS

The source range monitors provide the operator with information of the status of the neutron level in the core at very low power levels during startup and shutdown. At these power levels, reactivity additions shall not be made without this flux level information available to the operator. When the intermediate range monitors are on scale, adequate information is available without the SRMs and they can be retracted.

3/4.3.7.7 TRAVERSING IN-CORE PROBE SYSTEM

The OPERABILITY of the traversing in-core probe system with the specified minimum complement of equipment ensures that the measurements obtained from use of this equipment accurately represent the spatial neutron flux distribution of the reactor core.

MONITORING INSTRUMENTATION (Continued)

3/4.3.7.8 LOOSE-PART DETECTION SYSTEM

The OPERABILITY of the loose-part detection system ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.7.9 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent monitoring instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.7.10 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent monitoring instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM. This will ensure the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring and controlling the concentrations of potentially explosive gas mixtures in the main condenser offgas treatment system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

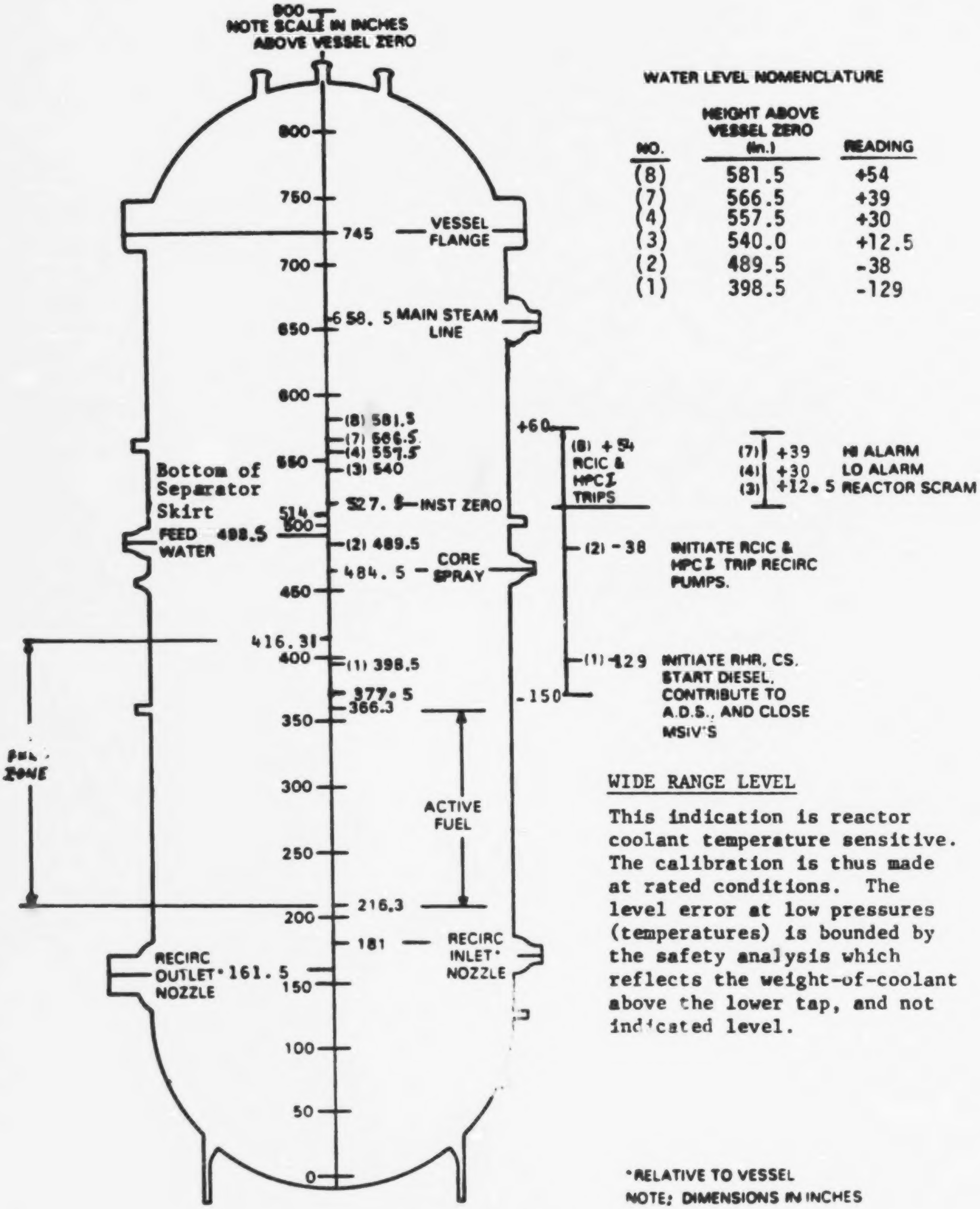
BEST COPY

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM

This specification is provided to ensure that the turbine overspeed protection system instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment or structures.

3/4.3.9 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

The feedwater/main turbine trip system actuation instrumentation is provided to initiate action of the feedwater system/main turbine trip system in the event of a high reactor vessel water level (Level 8) to mitigate potential damage to the main turbine.


900
NOTE SCALE IN INCHES ABOVE VESSEL ZERO
800
750
745
VESSEL FLANGE
700
658.5
MAIN STEAM LINE
650
600
(8) 581.5
(7) 566.5
(4) 557.5
550
(3) 540
Bottom of Separator Skirt
514
527.5
INST ZERO
500
FEED WATER
498.5
(2) 489.5
484.5
CORE SPRAY
450
416.31
400
(1) 398.5
377.5
366.3
350
FUEL ZONE
300
ACTIVE FUEL
250
216.3
200
181
RECIRC INLET* NOZZLE
RECIRC OUTLET* NOZZLE
161.5
150
100
50
0
+60
(8) +54
RCIC & HPCI TRIPS
(2) -38
INITIATE RCIC & HPCI TRIP RECIRC PUMPS.
(1) -129
INITIATE RHR, CS, START DIESEL, CONTRIBUTE TO A.D.S., AND CLOSE MSIV'S
-150
(7) +39 HI ALARM
(4) +30 LO ALARM
(3) +12.5 REACTOR SCRAM

WATER LEVEL NOMENCLATURE

NO.	HEIGHT ABOVE VESSEL ZERO (in.)	READING
(8)	581.5	+54
(7)	566.5	+39
(4)	557.5	+30
(3)	540.0	+12.5
(2)	489.5	-38
(1)	398.5	-129

WIDE RANGE LEVEL

This indication is reactor coolant temperature sensitive. The calibration is thus made at rated conditions. The level error at low pressures (temperatures) is bounded by the safety analysis which reflects the weight-of-coolant above the lower tap, and not indicated level.

*RELATIVE TO VESSEL
NOTE; DIMENSIONS IN INCHES

Bases Figure B 3/4 3-1

REACTOR VESSEL WATER LEVEL

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 RECIRCULATION SYSTEM

Operation with one reactor core coolant recirculation loop inoperable is prohibited until an evaluation of the performance of the ECCS during one loop operation has been performed, evaluated and determined to be acceptable.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design-basis-accident, increase the blowdown area and reduce the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation.

Recirculation pump speed mismatch limits are in compliance with the ECCS LOCA analysis design criteria. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than 145°F.

The objective of GE BWR plant and fuel design is to provide stable operation with margin over the normal operating domain. However, at the high power/low flow corner of the operating domain, a small probability of limit cycle neutron flux oscillations exists depending on combinations of operating conditions (e.g., rod pattern, power shape). To provide assurance that neutron flux limit cycle oscillations are detected and suppressed, APRM and LPRM neutron flux noise levels should be monitored while operating in this region.

Stability tests at operating BWRs were reviewed to determine a generic region of the power/flow map in which surveillance of neutron flux noise levels should be performed. A conservation decay ratio of 0.6 was chosen as the bases for determining the generic region for surveillance to account for the plant to plant variability of decay ratio with core and fuel designs. This generic region has been determined to correspond to a core flow of less than or equal to 45% of rated core flow and a THERMAL POWER greater than that specified in Figure 3.4.1.1-1.

Plant specific calculations can be performed to determine an applicable region for monitoring neutron flux noise levels. In this case the degree of conservatism can be reduced since plant to plant variability would be eliminated. In this case, adequate margin will be assured by monitoring the region which has a decay ratio greater than or equal to 0.8.

415

Neutron flux noise limits are also established to ensure early detection of limit cycle neutron flux oscillations. BWR cores typically operate with neutron flux noise caused by random boiling and flow noise. Typical neutron flux noise levels of 1-12% of rated power (peak-to-peak) have been reported for the range of low to high recirculation loop flow during both single and dual recirculation loop operation. Neutron flux noise levels which significantly bound these values are considered in the thermal/mechanical design of GE BWR fuel and are found to be of negligible consequence. In addition, stability tests at operating BWRs have demonstrated that when stability related neutron flux limit cycle oscillations occur they result in peak-to-peak neutron flux limit cycles of 5-10 times the typical values. Therefore, actions taken to reduce neutron flux noise levels exceeding three (3) times the typical value are sufficient to ensure early detection of limit cycle neutron flux oscillations.

Typically, neutron flux noise levels show a gradual increase in absolute magnitude as core flow is increased (constant control rod pattern) with two reactor recirculation loops in operation. Therefore, the baseline neutron flux noise level obtained at a specific core flow can be applied over a range of core flows. To maintain a reasonable variation between the low flow and high flow end of the flow range, the range over which a specific baseline is applied should not exceed 20% of rated core flow with two recirculation loops in operation. Data from tests and operating plants indicate that a range of 20% of rated core flow will result in approximately a 50% increase in neutron flux noise level during operation with two recirculation loops. Baseline data should be taken near the maximum rod line at which the majority of operation will occur. However, baseline date taken at lower rod lines (i.e., lower power) will result in a conservative value since the neutron flux noise level is proportional to the power level at a given core flow.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves operates to prevent the reactor coolant system from being pressurized above the Safety Limit of 1375 psig in accordance with the ASME Code. A total of 13 OPERABLE safety/relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case transient.

Demonstration of the safety/relief valve lift settings will occur only during shutdown. The safety/relief valves will be removed and either set pressure tested or replaced with spares which have been previously set pressure tested and stored in accordance with manufacturer's recommendations at the specified frequency.

The low-low set system ensures that safety/relief valve discharges are minimized for a second opening of these valves, following any overpressure transient. This is achieved by automatically lowering the closing setpoint of two valves and lowering the opening setpoint of two valves following the initial opening. In this way, the frequency and magnitude of the containment blowdown duty cycle is substantially reduced. Sufficient redundancy is provided for the low-low set system such that failure of any one valve to open or close at its reduced setpoint does not violate the design basis.

416

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so a 0.5 ppm concentration of chlorides is not considered harmful during these periods.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.5 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed small fractions of the dose guidelines of 10 CFR 100. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Monitoring the iodine activity in the primary coolant and taking responsible actions to maintain it at a reasonably low level will aid in ensuring the accumulated time of plant operation with high iodine activity will not exceed 800 hours in a consecutive 12-month period. The results of all primary coolant specific activity analyses which exceed the limits of Specification 3.4.5 will be documented pursuant to Specification 6.9.1.5.

Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analysis following power changes may be permissible if justified by the data obtained.

Closing the main steam line isolation valves prevents the release of activity to the environs should a steam line rupture occur outside containment. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section (4.9) of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

The operating limit curves of Figure 3.4.6.1-1 are derived from the fracture toughness requirements of 10 CFR 50 Appendix G and ASME Code Section III, Appendix G. The curves are based on the RT_{NDT} and stress intensity factor information for the reactor vessel components. Fracture toughness limits and the basis for compliance are more fully discussed in FSAR Chapter 5, Paragraph 5.3.1.5, "Fracture Toughness."

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 MeV, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, phosphorus content and copper content of the material in question, can be predicted using Bases Figure B 3/4.4.6-1 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The pressure/ temperature limit curve, Figure 3.4.6.1-1, curves A', B' and C', includes an assumed shift in RT_{NDT} for the end of life fluence.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, irradiated flux wires installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the flux wires and vessel inside radius are essentially identical, the irradiated flux wires can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.4.6.1-1 shall be adjusted, as required, on the basis of the flux wire data and recommendations of Regulatory Guide 1.99, Revision 1.

PRESSURE/TEMPERATURE LIMITS (Continued)

The pressure-temperature limit lines shown in Figures 3.4.6.1-1, curves C, and C', and A and A', for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance capsules and the frequencies for removing and testing the specimens in these capsules are provided in Table 4.4.6.1.3-1 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment, however, single failure considerations require that two valves be OPERABLE. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1977 Edition and Addenda through Summer 1978.

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.4.9 RESIDUAL HEAT REMOVAL

A single shutdown cooling mode loop provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication, however, single failure considerations require that two loops be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation.

BASES TABLE B 3/4.4.6-1

REACTOR VESSEL TOUGHNESS

BELTLINE COMPONENT	WELD SEAM I.D. OR MAT'L TYPE	HEAT/SLAB OR HEAT/LOT	CU(%)	P(%)	HIGHEST RT_{NDT}(°F)	PREDICTED Δ RT_{NDT}(°F)	UNIRRADIATED UPPER SHELF (FT-LBS)	MAX. EOL RT_{NDT}(°F)
Plate	SA-533 GR B CL.1	5K3025-1	.15	.012	+19	20	76	+39
Weld	Long. seams for shells 4&5 and girth weld between 4&5	D55040/ 1125-02000	.08	.010	-30	17	135	-13

NOTE: * These values are given only for the benefit of calculating the end-of-life (EOL) RT_{NDT}.

NON-BELTLINE COMPONENT	MT'L TYPE OR WELD SEAM I.D.	HEAT/SLAB OR HEAT/LOT	HIGHEST REFERENCE TEMPERATURE RT_{NDT} (°F)
Shell Ring Connected to Vessel Flange	SA 533, GR.B, Cl.1	All Heats	+19
Bottom Head Dome	SA 533, GR.B, Cl.1	All Heats	+30
Bottom Head Torus	SA 533, GR.B, Cl.1	All Heats	+30
LPCI Nozzles	SA 508, Cl.2	All Heats	-20
Top Head Torus	SA 533, GR.B, Cl.1	All Heats	+19
Top Head Flange	SA 508, Cl.2	All Heats	+10
Vessel Flange	SA 508, Cl.2	All Heats	+10
Feedwater Nozzle	SA 508, Cl.2	All Heats	-20
Weld Metal	All RPV Welds	All Heats	0
Closure Studs	SA 540, GR.B, 24	All Heats	Meet 45 ft-lbs & 25 mils lateral expansion at +10°F

The design of the Hope Creek vessel results in these nozzles experiencing a predicted EOL fluence at 1/4T of the vessel thickness of 1.6×10^{17} n/cm^2. Therefore, these nozzles are predicted to have an EOL RT_{NDT} of -6°F.

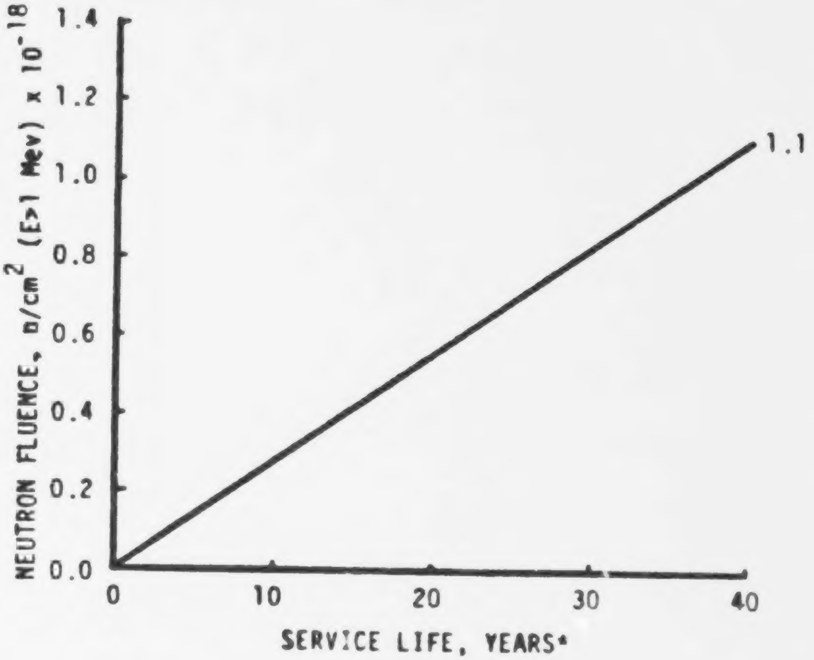


FIGURE B 3/4 4.6-1 FAST NEUTRON FLUENCE (E>1 Mev) AT 1/4 T AS A FUNCTION OF SERVICE LIFE*

Bases Figure B 3/4.4.6-1

---

* At 90% of RATED THERMAL POWER and 90% availability

422

3/4.5 EMERGENCY CORE COOLING SYSTEM

BASES

3/4.5.1 and 3/4.5.2 ECCS - OPERATING and SHUTDOWN

The core spray system (CSS), together with the LPCI mode of the RHR system, is provided to assure that the core is adequately cooled following a loss-of-coolant accident and provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the ADS.

The CSS is a primary source of emergency core cooling after the reactor vessel is depressurized and a source for flooding of the core in case of accidental draining.

The surveillance requirements provide adequate assurance that the CSS will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The low pressure coolant injection (LPCI) mode of the RHR system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. Four subsystems, each with one pump, provide adequate core flooding for all break sizes up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS.

The surveillance requirements provide adequate assurance that the LPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The high pressure coolant injection (HPCI) system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCI system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCI system continues to operate until reactor vessel pressure is below the pressure at which CSS operation or LPCI mode of the RHR system operation maintains core cooling.

The capacity of the system is selected to provide the required core cooling. The HPCI pump is designed to deliver greater than or equal to 5600 gpm at reactor pressures between 1120 and 200 psig. Initially, water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor, but no credit is taken in the safety analyses for the condensate storage tank water.

423

ECCS-OPERATING and SHUTDOWN (Continued)

With the HPCI system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the CSS and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, a system for which no credit is taken in the safety analysis, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCI out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems and the RCIC system.

The surveillance requirements provide adequate assurance that the HPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test with reactor vessel injection requires reactor to be in HOT SHUTDOWN with vessel pressure not less than 200 psig. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCI system to function properly after a small break loss-of-coolant accident, the automatic depressurization system (ADS) automatically causes selected safety-relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 100 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring ADS.

ADS automatically controls five selected safety-relief valves although the safety analysis only takes credit for four valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability.

3/4.5.3 SUPPRESSION CHAMBER

The suppression chamber is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCI, CSS and LPCI systems in the event of a LOCA. This limit on suppression chamber minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression chamber in OPERATIONAL CONDITIONS 1, 2 or 3 is also required by Specification 3.6.2.1.

Repair work might require making the suppression chamber inoperable. This specification will permit those repairs to be made and at the same time give assurance that the irradiated fuel has an adequate cooling water supply when the suppression chamber must be made inoperable, including draining, in OPERATIONAL CONDITION 4 or 5.

In OPERATIONAL CONDITION 4 and 5 the suppression chamber minimum required water volume is reduced because the reactor coolant is maintained at or below 200°F. Since pressure suppression is not required below 212°F, the minimum water volume is based on NPSH, recirculation volume and vortex prevention plus a safety margin for conservatism.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 48.1 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix "J" of 10 CFR Part 50 with the exception of exemptions granted for main steam isolation valve leak testing and testing the airlocks after each opening.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment.

3/4.6.1.4 MSIV SEALING SYSTEM

Calculated doses resulting from the maximum leakage allowance for the main steamline isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR 100 guidelines, provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIV's such that the specified leakage requirements have not always been maintained continuously. The sealing system will reduce the untreated leakage from the MSIVs when isolation of the primary system and containment is required.

3/4.6.1.5 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 48.1 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations on drywell and suppression chamber internal pressure ensure that the containment peak pressure of 48.1 psig does not exceed the design pressure of 62 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of 3 psid. The limit of -0.5 to +1.5 psig for initial positive containment pressure will limit the total pressure to 48.1 psig which is less than the design pressure and is consistent with the safety analysis.

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during LOCA conditions and is consistent with the safety analysis. The 135°F average temperature is conducive to normal and long term operation.

3/4.6.1.8 DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

The outboard 26-inch and outboard 24-inch drywell and suppression chamber purge supply and exhaust isolation valves are required to be sealed closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves sealed closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system. To provide assurance that the 26-inch and the 24-inch valves cannot be inadvertently opened, they are sealed closed in accordance with Standard Review Plan 6.2.4, which includes mechanical devices to seal or lock the valve closed, or prevent power from being supplied to the valve operator.

DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM (Continued)

The use of the drywell and suppression chamber purge lines for pressure control is restricted with the following exception, the inboard 26-inch valve on the drywell purge outlet vent line when used in conjunction with the 2-inch purge outlet vent line bypass valve since the 2-inch valves will close during a LOCA or steam line break accident and therefore the site boundary dose guidelines of 10 CFR Part 100 would not be exceeded in the event of an accident during purging operations. In addition due to the limited flow rate through the 2-inch bypass valve, the inboard 26-inch valve is also capable of closing under these conditions. The design of the 2-inch purge supply and exhaust isolation valves meets the requirements of Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations".

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2. DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure of 62 psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1020 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss of coolant accident, the pressure of the liquid must not exceed 62 psig, the suppression chamber maximum internal design pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant to be considered is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in this specification, containment pressure during the design basis accident is approximately 48.1 psig which is below the design pressure of 62 psig. Maximum water volume of 122,000 ft^3 results in a downcomer submergence of 3.33 ft and the minimum volume of 118,000 ft^3 results in a submergence of approximately 3.0 ft. The majority of the Bodega tests were run with a submerged length of four feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate. The maximum temperature at the end of the blowdown

DEPRESSURIZATION SYSTEMS (Continued)

tested during the Humboldt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Should it be necessary to make the suppression chamber inoperable, this shall only be done as specified in Specification 3.5.3.

Under full power operating conditions, blowdown from an initial suppression chamber water temperature of 95°F results in a water temperature of approximately 135°F immediately following blowdown which is below the 200°F used for complete condensation via mitered T-quencher devices. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is no dependency on containment overpressure during the accident injection phase. If both RHR loops are used for containment cooling, there is no dependency on containment overpressure for post-LOCA operations.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak local temperature of the suppression pool is maintained below 200°F during any period of relief valve operation. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety-relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety-relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety relief valve to assure mixing and uniformity of energy insertion to the pool.

In conjuction with the Mark I containment Long Term Program, a plant unique analysis was performed which demonstrated that the containment, the attached piping and internal structures meet the applicable structural and mechanical acceptance criteria for Hope Creek. The evaluation followed the design basis loads defined in the Mark I Load Definition Report, NEDO-21888, December 1978, as modified by NRC SER NUREG 0661, July 1980 and Supplement 1, August 1982, to ensure that hydrodynamic loads, appropriate for the life of the plant, were applied.

BEST COPY

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through 57 of Appendix A of 10 CFR 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 VACUUM RELIEF

Vacuum relief breakers are provided to equalize the pressure between the suppression chamber and drywell and between the Reactor Building and suppression chamber. This system will maintain the structural integrity of the primary containment under conditions of large differential pressures.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident.

3/4.6.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The Reactor Building and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified.

Establishing and maintaining a 0.25 inch water gage vacuum in the reactor building with the filtration recirculation and ventilation system (FRVS) once per 18 months, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

The OPERABILITY of the FRVS ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses and with the drawdown analysis. Continuous operation of the system with the heaters and humidity control instruments OPERABLE for 10 hours during each 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL

The OPERABILITY of the systems required for the detection and control of hydrogen gas ensures that these systems will be available to maintain the hydrogen concentration within the primary containment below its flammable limit during post-LOCA conditions. Either containment hydrogen recombiner is capable of controlling the expected hydrogen generation associated with (1) zirconium-water reactions, (2) radiolytic decomposition of water and (3) corrosion of metals within containment. The hydrogen control system is consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA" November 1978.

3/4.7.1 SERVICE WATER SYSTEMS

The OPERABILITY of the station service water and the safety auxiliaries cooling systems ensures that sufficient cooling capacity is available for continued operation of the SACS and its associated safety-related equipment during normal and accident conditions. The redundant cooling capacity of these systems, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

3/4.7.2 CONTROL ROOM EMERGENCY FILTRATION SYSTEM

The OPERABILITY of the control room emergency filtration system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all design basis accident conditions. Continuous operation of the system with the heaters and humidity control instruments OPERABLE for 10 hours during each 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR Part 50.

3/4.7.3 FLOOD PROTECTION

The requirement for flood protection ensures that facility flood protection features are in place in the event of flood conditions. The limit of elevation 10.5' Mean Sea Level is based on the elevation at which facility flood protection features provide protection to safety related equipment.

3/4.7.4 REACTOR CORE ISOLATION COOLING SYSTEM

The reactor core isolation cooling (RCIC) system is provided to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without requiring actuation of any of the Emergency Core Cooling System equipment. The RCIC system is conservatively required to be OPERABLE whenever reactor steam dome pressure exceeds 150 psig. This pressure is substantially below that for which the RCIC system can provide adequate core cooling for events requiring the RCIC system.

The RCIC system specifications are applicable during OPERATIONAL CONDITIONS 1, 2 and 3 when reactor vessel steam dome pressure exceeds 150 psig because RCIC is the primary non-ECCS source of emergency core cooling when the reactor is pressurized.

With the RCIC system inoperable, adequate core cooling is assured by the OPERABILITY of the HPCI system and justifies the specified 14 day out-of-service period.

REACTOR CORE ISOLATION COOLING SYSTEM (Continued)

The surveillance requirements provide adequate assurance that RCIC will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to start cooling at the earliest possible moment.

3/4.7.5 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety related system.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip, and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the Plant Operations Review Committee. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guide 8.8 and 8.10. The addition or deletion of any snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to each safety-related system. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. In order to establish the inspection frequency for each type of snubber on a safety-related system, it was assumed that the frequency of snubber failures and initiating events is constant with time and that the failure of any snubber on that system could cause the system to be unprotected and to result in failure during an assumed initiating event. Inspections performed before that

BEST CO

SNUBBERS (Continued)

interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elasped (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results required a shorter inspection interval will override the previous schedule.

The acceptance criteria are to be used in the visual inspection to determine OPERABILITY of the snubbers.

To provide assurance of snubber functional reliability one of three functional testing methods is used with the stated acceptance criteria:

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or
2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.7.5-1, or
3. Functionally test a representative sample size and determine sample acceptance or rejection using the stated equation.

Figure 4.7.5-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in Quality Control and Industrial Statistics" by Acheson J. Duncan.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubbers for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in the list of individual snubbers indicating the extent of the exemptions.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (i.e., newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

3/4.7.6 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism, i.e., sealed sources within radiation monitoring devices, are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.7 MAIN TURBINE BYPASS SYSTEM

The main turbine bypass system is required to be OPERABLE consistent with the assumptions of the feedwater controller failure analysis for FSAR Chapter 15.

3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for (1) the safe shutdown of the facility and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criteria 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one of the onsite A.C. and the corresponding D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. or D.C. source.

The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources", December 1974 as modified by plant specific analysis and diesel generator manufacturer recommendations. When two diesel generators are inoperable, there is an additional ACTION requirement to verify that all required systems, susbsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generators as a source of emergency power, are also OPERABLE. This requirement is intended to provide assurance that a loss of offsite power event will not result in a complete loss of safety function of critical systems during the period two or more of the diesel generators are inoperable. The term verify as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the component.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that (1) the facility can be maintained in the shutdown or refueling condition for extended time periods and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The surveillance requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guide 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies", March 10, 1971, Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants", Revision 1, August 1977 and Regulatory Guide 1.137" Fuel-Oil Systems for Standby Diesel Generators", Revision 1, October 1979 as modified by plant specific analysis and diesel generator manufacturer's recommendations.

435

A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

The surveillance requirements for demonstrating the OPERABILITY of the unit batteries are in accordance with the recommendations of Regulatory Guide 1.129 "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants", February 1978 and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8.2.1-1 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and .015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than .020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8.2.1-1 is permitted for up to 7 days. During this 7 day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than .020 below the manufacturer's recommended full charge specific gravity ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than .040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

436

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Primary containment electrical penetrations and penetration conductors are protected by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers by periodic surveillance.

The surveillance requirements applicable to lower voltage circuit breakers provides assurance of breaker reliability by testing at least one representative sample of each manufacturers brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are than tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.

The OPERABILITY or bypassing of the motor operated valves thermal overload protection continuously or during accident conditions by integral bypass devices ensures that the thermal overload protection during accident conditions will not prevent safety related valves from performing their function. The Surveillance Requirements for demonstrating the OPERABILITY or bypassing of the thermal overload protection continuously or during accident conditions are in accordance with Regulatory Guide 1.106 "Thermal Overload Protection for Electric Motors or Motor Operated Valves", Revision 1, March 1977.

3/4.9.1 REACTOR MODE SWITCH

Locking the OPERABLE reactor mode switch in the Shutdown or Refuel position, as specified, ensures that the restrictions on control rod withdrawal and refueling platform movement during the refueling operations are properly activated. These conditions reinforce the refueling procedures and reduce the probability of inadvertent criticality, damage to reactor internals or fuel assemblies, and exposure of personnel to excessive radiation.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of at least two source range monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 CONTROL ROD POSITION

The requirement that all control rods be inserted during other CORE ALTERATIONS minimizes the possibility that fuel will be loaded into a cell without a control rod, although one rod may be withdrawn under control of the reactor mode switch refuel position one-rod-out-interlock.

3/4.9.4 DECAY TIME

The minimum requirement for reactor subcriticality prior to fuel movement ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during movement of fuel within the reactor pressure vessel.

3/4.9.6 REFUELING PLATFORM

The OPERABILITY requirements ensure that (1) the refueling platform will be used for handling control rods and fuel assemblies within the reactor pressure vessel, (2) each crane and hoist has sufficient load capacity for handling fuel assemblies and control rods, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL

The restriction on movement of loads in excess of the nominal weight of a fuel assembly over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 and 3/4.9.9 WATER LEVEL - REACTOR VESSEL and WATER LEVEL - SPENT FUEL STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.10 CONTROL ROD REMOVAL

These specifications ensure that maintenance or repair of control rods or control rod drives will be performed under conditions that limit the probability of inadvertent criticality. The requirements for simultaneous removal of more than one control rod are more stringent since the SHUTDOWN MARGIN specification provides for the core to remain subcritical with only one control rod fully withdrawn.

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal loop be OPERABLE or that an alternate method capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during REFUELING, and (2) sufficient coolant circulation would be available through the reactor core to assure accurate temperature indication and to distribute and prevent stratification of the poison in the event it becomes necessary to actuate the standby liquid control system.

The requirement to have two shutdown cooling mode loops OPERABLE when there is less than 22 feet 2 inches of water above the reactor vessel flange ensures that a single failure of the operating loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 22 feet 2 inches of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event a failure of the operating RHR loop, adequate time is provided to initiate alternate methods capable of decay heat removal or emergency procedures to cool the core.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

The requirement for PRIMARY CONTAINMENT INTEGRITY is not applicable during the period when open vessel tests are being performed during the low power PHYSICS TESTS.

3/4.10.2 ROD SEQUENCE CONTROL SYSTEM

In order to perform the tests required in the technical specifications it is necessary to bypass the sequence restraints on control rod movement. The additional surveillance requirments ensure that the specifications on heat generation rates and shutdown margin requirements are not exceeded during the period when these tests are being performed and that individual rod worths do not exceed the values assumed in the safety analysis.

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

Performance of shutdown margin demonstrations during open vessel testing requires additional restrictions in order to ensure that criticality is properly monitored and controlled. These additional restrictions are specified in this LCO.

3/4.10.4 RECIRCULATION LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 OXYGEN CONCENTRATION

Relief from the oxygen concentration specifications is necessary in order to provide access to the primary containment during the initial startup and testing phase of operation. Without this access the startup and test program could be restricted and delayed.

3/4.10.6 TRAINING STARTUPS

This special test exception permits training startups to be performed with the reactor vessel depressurized at low THERMAL POWER and temperature while controlling RCS temperature with one RHR subsystem aligned in the shutdown cooling mode in order to minimize contaminated water discharge to the radioactive waste disposal system.

3/4.10.7 SPECIAL INSTRUMENTATION - INITIAL CORE LOADING

This special test exception permits relief from the requirements for a minimum count rate while loading the first 16 fuel bundles to allow sufficient source-to-detector coupling such that minimum count rate can be achieved on an SRM. This is acceptable because of the significant margin to criticality while loading the initial 16 fuel bundles.

3/4.11 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in Currie, L. A., "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," NUREG/CR-4007 (September 1984), and in the HASL Procedures Manual, HASL-300 (revised annually).

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR Part 141. The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

3/4.11.1.3 LIQUID RADWASTE TREATMENT SYSTEM

The OPERABILITY of the liquid radwaste treatment system ensures that this system will be available for use whenever liquid effluents require treatment prior to their release to the environment. The requirement that the appropriate portions of this system be used, when specified, provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks listed in this specification include all those outdoor radwaste tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose at any time at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 to UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. Examples of calculations for such MEMBERS OF THE PUBLIC, with the appropriate occupancy factors, shall be given in the ODCM. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrems/year to the total body or to less than or equal to 3000 mrems/year to the skin.

DOSE RATE (Continued)

These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in Currie, L. A., "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," NUREG/CR-4007 (September 1984), and in the HASL Procedures Manual, HASL-300 (revised annually).

3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

3/4.11.2.3 DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements implement

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM (Continued)

the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half lives greater than 8 days are dependent upon the existing radionuclide pathways to man, in the areas at and beyond the SITE BOUNDARY. The pathways that were examined in the development of these calculations were: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

3/4.11.2.4 AND 3/4.11.2.5 GASEOUS RADWASTE TREATMENT AND VENTILATION EXHAUST TREATMENT

The OPERABILITY of the GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the system will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

3/4.11.2.6 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the GASEOUS RADWASTE TREATMENT SYSTEM main condenser offgas system is maintained below the flammability limits of hydrogen and oxygen. Automatic control features are included in the system to prevent the hydrogen and oxygen concentration from reaching these flammability

EXPLOSIVE GAS MIXTURE (Continued)

limits. These automatic control features include isolation of the source of hydrogen and/or oxygen on loss of dilution steam. Maintaining the concentration of hydrogen below the flammability limit provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.7 MAIN CONDENSER

Restricting the gross radioactivity rate of noble gases from the main condenser provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10 CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50.

3/4.11.2.8 VENTING OR PURGING

This specification provides reasonable assurance that releases from drywell purging operations will not exceed the annual dose limits of 10 CFR Part 20 for UNRESTRICTED AREAS.

3/4.11.3 SOLID RADIOACTIVE WASTE TREATMENT

This specification implements the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, and mixing and curing times. The purpose of the PROCESS CONTROL PROGRAM is to provide quality assurance that the solidified waste meets 10 CFR Part 61 requirements.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant generated radioactive effluents and direct radiation exceed 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the reactor units including outside storage tanks,etc. are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF

TOTAL DOSE (Continued)

THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR Part 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.11.1.1 and 3.11.2.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

3/4.12.1 MONITORING PROGRAM

The radiological environmental monitoring program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring, Revision 1, November 1979. The initially specified monitoring program will be effective for at least the first 3 years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.12.1-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in Currie, L. A., "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," NUREG/CR-4007 (September 1984), and in the HASL Procedures Manual, HASL-300 (revised annually).

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the radiological environmental monitoring program are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey, from visual survey or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 50 m^2 provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: (1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and (2) a vegetation yield of 2 kg/m^2.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

448

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA AND MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.1 The exclusion area shall be as shown in Figure 5.1.1-1. Information regarding radioactive gaseous and liquid effluents which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be as shown in Figure 5.1.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1.2-1. The circle with the five mile radius is the low population zone.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The primary containment is a steel structure composed of a spherical lower portion, a cylindrical middle portion, and a hemispherical top head which form a drywell. The drywell is attached to the suppression chamber through a series of downcomer vents. The suppression chamber is a steel pressure vessel in the shape of a torus. The drywell has a nominal free air volume of 169,000 cubic feet. The suppression chamber has an air volume of 137,000 cubic feet and a water region of 118,000 cubic feet.

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The primary containment is designed and shall be maintained for:

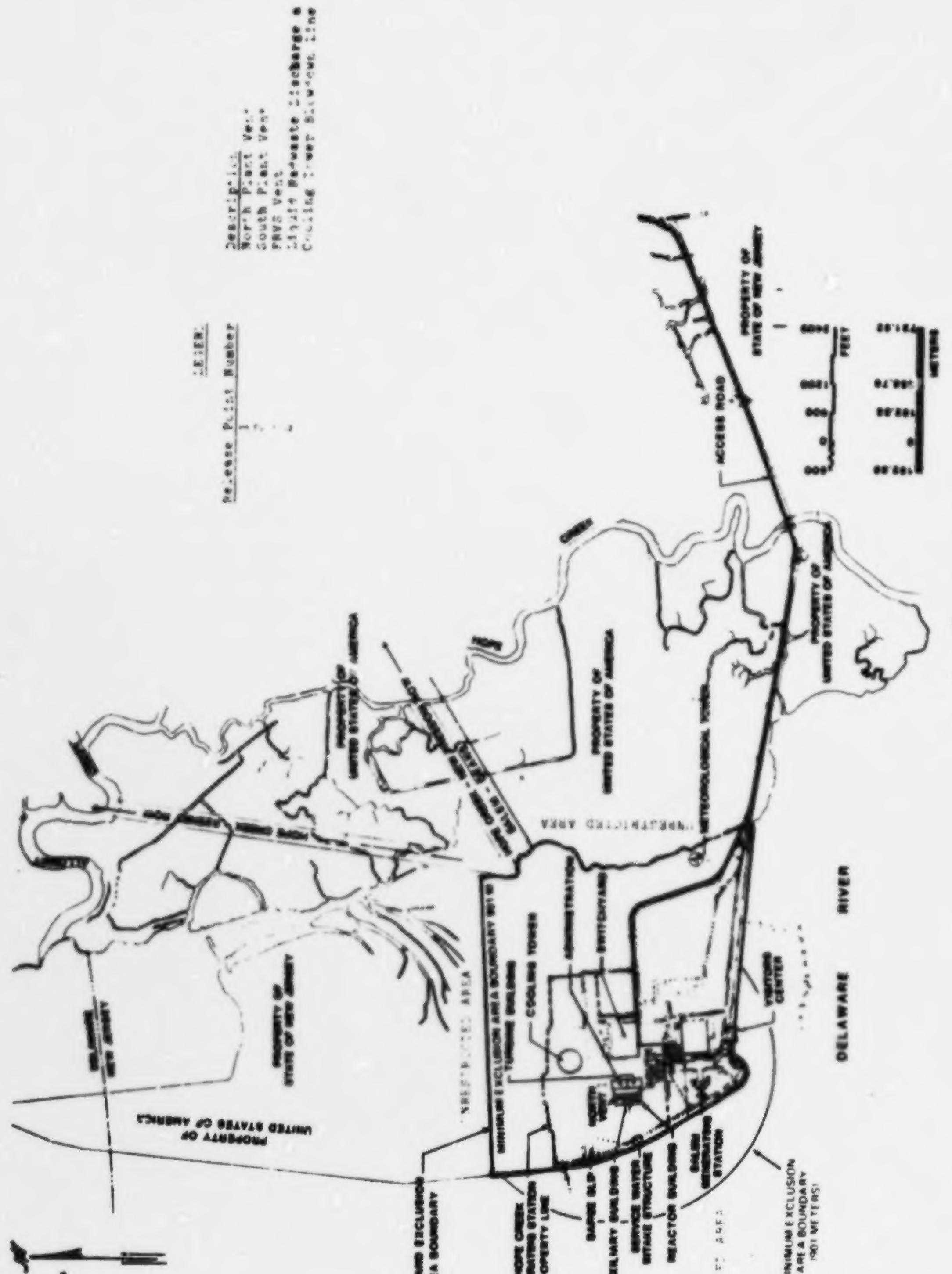
a. Maximum internal pressure 62 psig.

b. Maximum internal temperature: drywell 340°F.
suppression pool 310°F.

c. Maximum external differential pressure 3 psid.

SECONDARY CONTAINMENT

5.2.3 The secondary containment consists of the Reactor Building, and a portion of the main steam tunnel and has a free volume of 4,000,000 cubic feet.

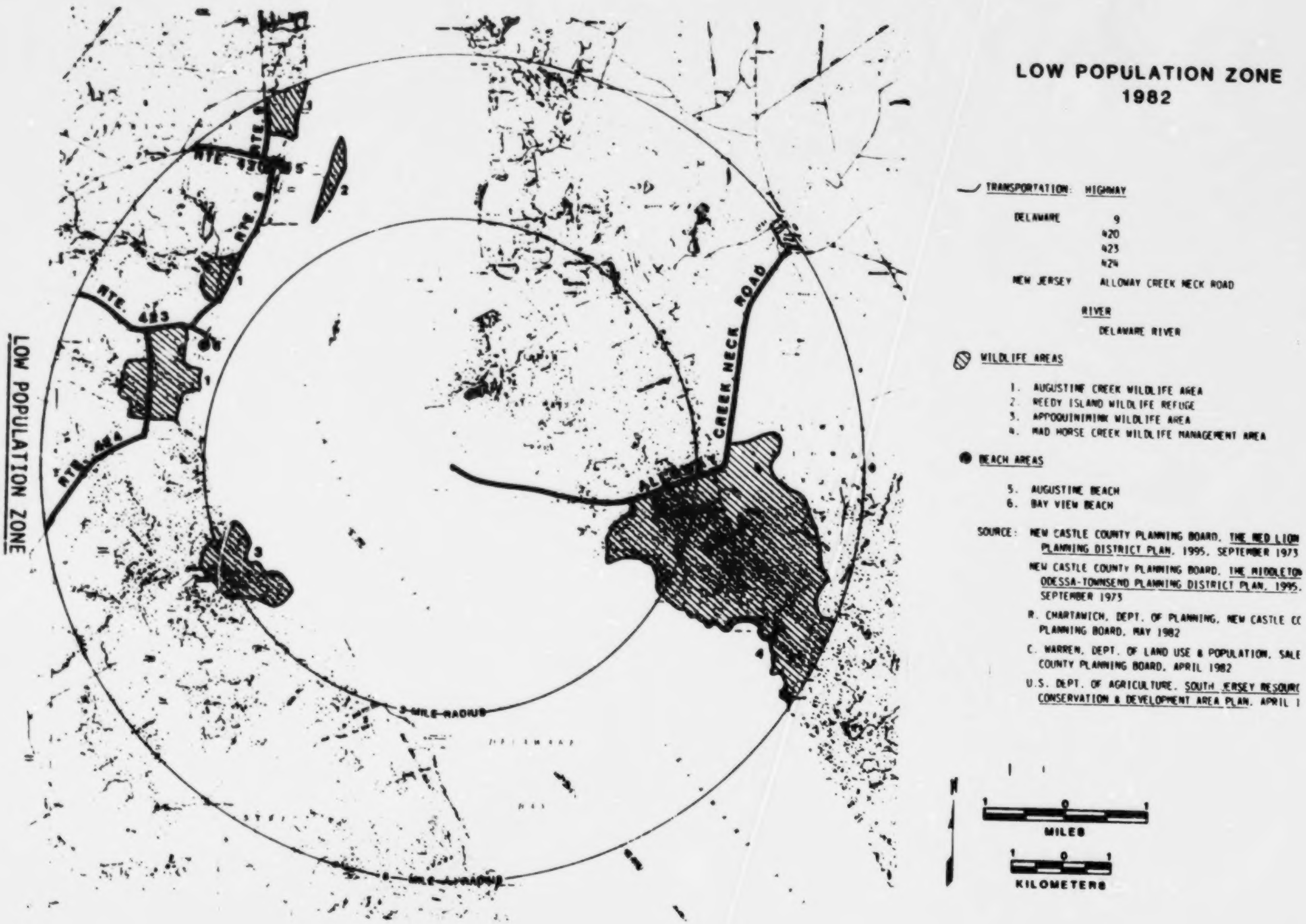

LEGEND:
Release Point Number
Description
1 North Plant Vent
2 South Plant Vent
3 FRVS Vent
4 Liquid Radwaste Discharge & Cooling Tower Blowdown Line
DELAWARE
NEW JERSEY
PROPERTY OF UNITED STATES OF AMERICA
PROPERTY OF STATE OF NEW JERSEY
PROPERTY OF UNITED STATES OF AMERICA
PROPERTY OF UNITED STATES OF AMERICA
PROPERTY OF UNITED STATES OF AMERICA
PROPERTY OF STATE OF NEW JERSEY
HOPE CREEK
SITE AND EXCLUSION AREA BOUNDARY
UNRESTRICTED AREA
MINIMUM EXCLUSION AREA BOUNDARY 901 M
TURBINE BUILDING
COOLING TOWER
ADMINISTRATION
SWITCHYARD
HOPE CREEK GENERATING STATION PROPERTY LINE
BARGE SLIP
NORTH VENT
AUXILIARY BUILDING
SERVICE WATER INTAKE STRUCTURE
REACTOR BUILDING
SALEM GENERATING STATION
UNRESTRICTED AREA
VISITORS CENTER
MINIMUM EXCLUSION AREA BOUNDARY (901 METERS)
UNRESTRICTED AREA
METEOROLOGICAL TOWER
ACCESS ROAD
DELAWARE RIVER
DELAWARE RIVER
600 0 600 1200 2400
FEET
182.88 0 182.88 365.76 731.52
METERS

450

EXCLUSION AREA AND UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

FIGURE 5.1.1-1

BEST COPY AVAILABLE


RTE. 9
RTE. 420
RTE. 423
RTE. 424
ALLOWAY CREEK NECK ROAD
1
2
3
4
5
6
2-MILE RADIUS
5-MILE RADIUS
DELAWARE
BAY
N
1 0 1
MILES
1 0 1
KILOMETERS

LOW POPULATION ZONE
1982

TRANSPORTATION: HIGHWAY

DELAWARE	9
	420
	423
	424
NEW JERSEY	ALLOWAY CREEK NECK ROAD

RIVER

DELAWARE RIVER

▨ **WILDLIFE AREAS**

1. AUGUSTINE CREEK WILDLIFE AREA
2. REEDY ISLAND WILDLIFE REFUGE
3. APPOQUINIMINK WILDLIFE AREA
4. MAD HORSE CREEK WILDLIFE MANAGEMENT AREA

● **BEACH AREAS**

5. AUGUSTINE BEACH
6. BAY VIEW BEACH

SOURCE: NEW CASTLE COUNTY PLANNING BOARD, THE RED LION PLANNING DISTRICT PLAN, 1995, SEPTEMBER 1973

NEW CASTLE COUNTY PLANNING BOARD, THE MIDDLETON ODESSA-TOWNSEND PLANNING DISTRICT PLAN, 1995, SEPTEMBER 1973

R. CHARTAWICH, DEPT. OF PLANNING, NEW CASTLE CC PLANNING BOARD, MAY 1982

C. WARREN, DEPT. OF LAND USE & POPULATION, SALE COUNTY PLANNING BOARD, APRIL 1982

U.S. DEPT. OF AGRICULTURE, SOUTH JERSEY RESOURC CONSERVATION & DEVELOPMENT AREA PLAN, APRIL 1

FIGURE 5.1.2-1

LOW POPULATION ZONE

451

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies with each fuel assembly containing 62 fuel rods and two water rods clad with Zircaloy-2. Each fuel rod shall have a nominal active fuel length of 150 inches. The initial core loading shall have a maximum average enrichment of 1.90 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 control rod assemblies, each consisting of a cruciform array of stainless steel tubes containing 143 inches of boron carbide, B_4C, powder surrounded by a cruciform shaped stainless steel sheath.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of:

   1. 1250 psig on the suction side of the recirculation pump.
   2. 1500 psig from the recirculation pump discharge to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 21,970 cubic feet at a nominal steam dome saturation temperature of 547°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1.1-1.

452

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, including all calculational uncertainties and biases as described in Section 9.1.2 of the FSAR.

b. A nominal 6.308 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 199' 4".

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1108 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7.1-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7.1-1.

TABLE 5.7.1-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor	120 heatup and cooldown cycles	70°F to 546°F to 70°F
	80 step change cycles	Loss of feedwater heaters
	180 reactor trip cycles	100% to 0% of RATED THERMAL POWER
	130 hydrostatic pressure and leak tests	Pressurized to $\geq$ 930 and $\leq$1250 psig

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The General Manager - Hope Creek Operations shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Senior Nuclear Shift Supervisor or during his absence from the control room, a designated individual shall be responsible for the control room command function. A management directive to this effect, signed by the Vice President - Nuclear shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown on Figure 6.2.1-1.

UNIT STAFF

6.2.2 The unit organization shall be as shown on Figure 6.2.2-1 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2.2-1;

b. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in OPERATIONAL CONDITION 1, 2 or 3, at least one licensed Senior Reactor Operator shall be in the control room;

c. A Radiation Protection Technician* shall be on site when fuel is in the reactor;

d. ALL CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or licensed Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation; and

---

*The Radiation Protection Technician may be unavailable for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required position.

455

UNIT STAFF (continued)

e. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions e.g., licensed Senior Reactor Operators, licensed Reactor Operators, radiation protection technicians, equipment operators, and key maintenance personnel.

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance on major unit modifications, on a temporary basis the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.
2. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7 day period, all excluding shift turnover time.
3. A break of at least 8 hours should be allowed between work periods, including shift turnover time.
4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the appropriate department manager, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the General Manager-Hope Creek Operations or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

BLANK PAGE

BLANK PAGE

BLANK PAGE

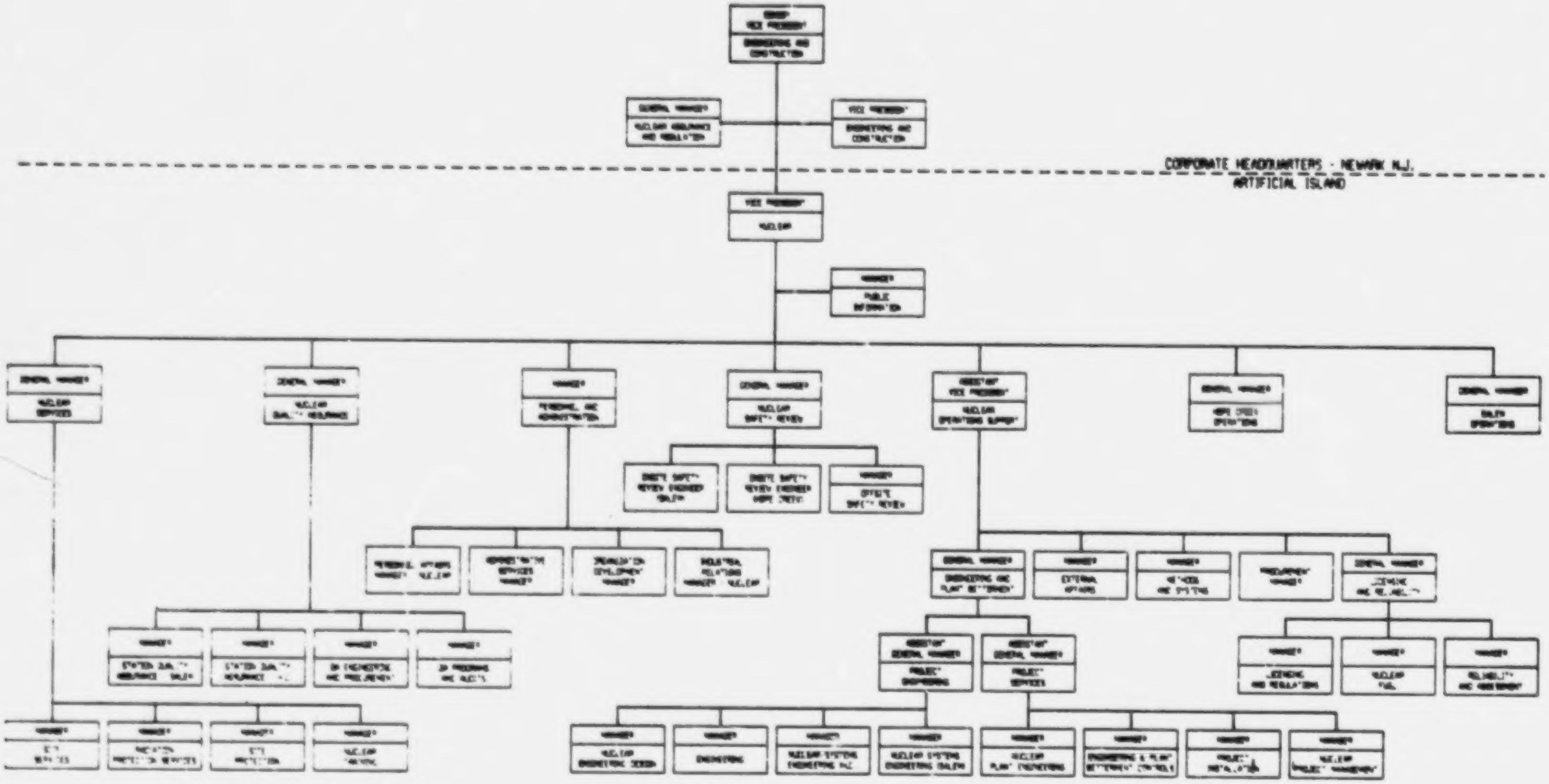

CORPORATE HEADQUARTERS - NEWARK N.J.
ARTIFICIAL ISLAND

FIGURE 6.2.1-1

OFFSITE ORGANIZATION

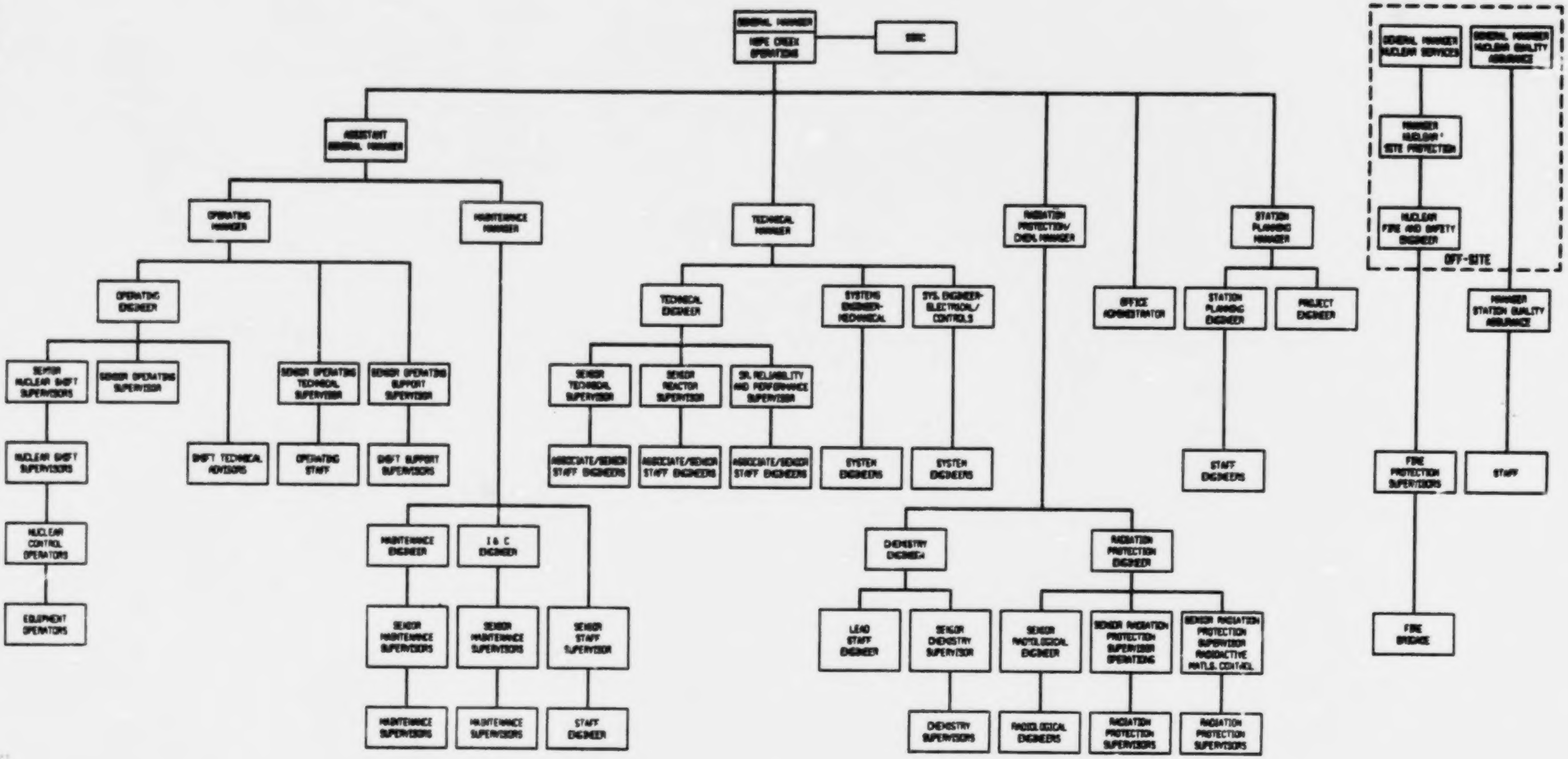

GENERAL MANAGER
HOPE CREEK OPERATIONS
SORC
ASSISTANT GENERAL MANAGER
OPERATING MANAGER
MAINTENANCE MANAGER
TECHNICAL MANAGER
RADIATION PROTECTION/ CHEM. MANAGER
OFFICE ADMINISTRATOR
STATION PLANNING MANAGER
GENERAL MANAGER NUCLEAR SERVICES
GENERAL MANAGER NUCLEAR QUALITY ASSURANCE
MANAGER NUCLEAR SITE PROTECTION
NUCLEAR FIRE AND SAFETY ENGINEER
OFF-SITE
OPERATING ENGINEER
SENIOR NUCLEAR SHIFT SUPERVISORS
SENIOR OPERATING SUPERVISOR
SHIFT TECHNICAL ADVISORS
SENIOR OPERATING TECHNICAL SUPERVISOR
SENIOR OPERATING SUPPORT SUPERVISOR
OPERATING STAFF
SHIFT SUPPORT SUPERVISORS
NUCLEAR SHIFT SUPERVISORS
NUCLEAR CONTROL OPERATORS
EQUIPMENT OPERATORS
MAINTENANCE ENGINEER
I & C ENGINEER
SENIOR MAINTENANCE SUPERVISORS
SENIOR MAINTENANCE SUPERVISORS
SENIOR STAFF SUPERVISOR
MAINTENANCE SUPERVISORS
MAINTENANCE SUPERVISORS
STAFF ENGINEER
TECHNICAL ENGINEER
SYSTEMS ENGINEER-MECHANICAL
SYS. ENGINEER-ELECTRICAL/ CONTROLS
SENIOR TECHNICAL SUPERVISOR
SENIOR REACTOR SUPERVISOR
SR. RELIABILITY AND PERFORMANCE SUPERVISOR
ASSOCIATE/SENIOR STAFF ENGINEERS
ASSOCIATE/SENIOR STAFF ENGINEERS
ASSOCIATE/SENIOR STAFF ENGINEERS
SYSTEM ENGINEERS
SYSTEM ENGINEERS
CHEMISTRY ENGINEER
RADIATION PROTECTION ENGINEER
LEAD STAFF ENGINEER
SENIOR CHEMISTRY SUPERVISOR
SENIOR RADIOLOGICAL ENGINEER
SENIOR RADIATION PROTECTION SUPERVISOR OPERATING
SENIOR RADIATION PROTECTION SUPERVISOR RADIOACTIVE MATLS. CONTROL
CHEMISTRY SUPERVISORS
RADIOLOGICAL ENGINEERS
RADIATION PROTECTION SUPERVISORS
RADIATION PROTECTION SUPERVISORS
STATION PLANNING ENGINEER
PROJECT ENGINEER
STAFF ENGINEERS
FIRE PROTECTION SUPERVISORS
FIRE BRIGADE
MANAGER STATION QUALITY ASSURANCE
STAFF

FIGURE 6.2.2-1

UNIT ORGANIZATION

TABLE 6.2.2-1

MINIMUM SHIFT CREW COMPOSITION

SINGLE UNIT FACILITY

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	CONDITION 1, 2, or 3	CONDITION 4 or 5
SNSS*	1	1
NSS*	1	None
NCO	2	1
EO	2	1
STA	1	None

TABLE NOTATION

SNSS - Senior Nuclear Shift Supervisor with a Senior Reactor Operator license on the Unit
NSS - Nuclear Shift Supervisor with a Senior Reactor Operator license on the Unit
NCO - Nuclear Control Operator with a Reactor Operator license on the Unit
EO - Equipment Operator
STA - Shift Technical Advisor

Except for the Senior Nuclear Shift Supervisor, the shift crew composition may be one less than the minimum requirements of Table 6.2.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Senior Nuclear Shift Supervisor from the control room while the unit is in OPERATIONAL CONDITION 1, 2 or 3, an individual with a valid Senior Reactor Operator license shall be designated to assume the control room command function. During any absence of the Senior Nuclear Shift Supervisor from the control room while the unit is in OPERATIONAL CONDITION 4 or 5, an individual with a valid Senior Reactor Operator license or Operator license shall be designated to assume the control room command function.

---

*In cases where an individual has a Senior Reactor Operator's license on the unit, is a qualified STA, and has a Professional Engineers License by virtue of successful completion of the Professional Engineers examination or a bachelor's degree in a scientific, engineering, or engineering technology discipline from an accredited institution, the individual can serve in a dual role capacity as either the SNSS/STA or NSS/STA. (Note: For those individuals with a bachelor's degree in a scientific or engineering technology discipline, course work must have included physical, mathematical, or engineering science.) Otherwise, there shall be a qualified STA as well as a SNSS and NSS on-shift.

6.2.3 SHIFT TECHNICAL ADVISOR

6.2.3.1 The Shift Technical Advisor shall provide advisory technical support to the Senior Nuclear Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to safe operation of the unit. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1981 for comparable positions, except for the Radiation Protection Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. The licensed Reactor Operators and Senior Reactor Operators shall also meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Manager-Nuclear Training, shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI/ANS 3.1-1981 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Manager - Site Protection and shall meet or exceed the requirements of the SRP (NUREG-0800) Section 13.2.2.II.6, 10 CFR 50 Appendix R and Branch Technical Position CMEB 9.5.1, Section C.3.d.

6.5 REVIEW AND AUDIT

6.5.1 STATION OPERATIONS REVIEW COMMITTEE (SORC)

FUNCTION

6.5.1.1 The SORC shall function to advise the General Manager - Hope Creek Operations on all matters related to nuclear safety.

460

BEST COPY

COMPOSITION

6.5.1.2 The SORC shall be composed of the:

Chairman:	General Manager - Hope Creek Operations
Member and Vice Chairman:	Operations Manager
Member and Vice Chairman:	Technical Manager
Member:	Maintenance Manager
Member:	I & C Engineer
Member:	Systems Engineer
Member:	Radiation Protection/Chemistry Manager
Member:	Radiation Protection Engineer or Chemistry Engineer
Member:	Onsite Safety Review Engineer (or designee)

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the SORC Chairman.

a. Vice Chairmen shall be members of Station management.

b. No more than two alternates to members shall participate as voting members in SORC activities at any one meeting.

c. Alternate appointees will only represent their respective department.

d. Alternates for members will not make up part of the voting quorum when the member the alternate represents is also present.

MEETING FREQUENCY

6.5.1.4 The SORC shall meet at least once per calendar month and as convened by the SORC Chairman or his designated alternate.

QUORUM

6.5.1.5 The quorum of the SORC necessary for the performance of the SORC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least four members including alternates.

RESPONSIBILITIES

6.5.1.6 The SORC shall be responsible for:

a. Review of: (1) all Station Administrative Procedures and changes thereto and (2) Newly created procedures or changes to existing procedures that involve a significant safety issue as described in Section 6.5.3.2.d.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to Appendix "A" Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Review of the safety evaluations that have been completed under the provisions of 10 CFR 50.59.

f. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluations and recommendations to prevent recurrence to the Vice President - Nuclear and to the General Manager - Nuclear Safety Review.

g. Review of all REPORTABLE EVENTS.

h. Review of facility operations to detect potential nuclear safety hazards.

i. Performance of special reviews, investigations or analyses and reports thereon as requested by the General Manager - Hope Creek Operations or General Manager - Nuclear Safety Review.

j. Review of the Facility Security Plan and implementing procedures and shall submit recommended changes to the General Manager - Nuclear Safety Review.

k. Review of the Facility Emergency Plan and implementing procedures and shall submit recommended changes to the General Manager - Nuclear Safety Review.

l. Review of the Fire Protection Program and implementing procedures and shall submit recommended changes to the General Manager - Nuclear Safety Review.

m. Review of all unplanned on-site releases of radioactivity to the environs including the preparation of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President - Nuclear and to the General Manager - Nuclear Safety Review.

n. Review of changes to the PROCESS CONTROL MANUAL and the OFF-SITE DOSE CALCULATION MANUAL, and the Radwaste Treatment Systems.

REVIEW PROCESS

6.5.1.7 A technical review and control system utilizing qualified reviewers shall function to perform the periodic or routine review of procedures and changes thereto. Details of this technical review process are provided in Section 6.5.3.

462

BEST COPY

AUTHORITY

6.5.1.8 The SORC shall:

a. Recommend in writing to the General Manager - Hope Creek Operations approval or disapproval of items considered under Specification 6.5.1.6.a. through e. prior to their implementation.

b. Provide written notification within 24 hours to the Vice President - Nuclear and to the General Manager - Nuclear Safety Review of disagreement between the SORC and the General Manager - Hope Creek Operations; however, the General Manager - Hope Creek Operations shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS

6.5.1.9 The SORC shall maintain minutes of each SORC meeting, and copies shall be provided to the Vice President - Nuclear, General Manager - Nuclear Safety Review and Manager - Offsite Safety Review.

6.5.2 NUCLEAR SAFETY REVIEW

FUNCTION

6.5.2.1 The Nuclear Safety Review Department (NSR) shall function to provide the independent safety review program and audit of designated activities.

COMPOSITION

6.5.2.2 NSR shall consist of the General Manager - Nuclear Safety Review, the Manager - Offsite Safety Review, who is supported by at least four dedicated, full-time engineers, and the Onsite Safety Review Group, which is managed by the Onsite Safety Review Engineer and is supported by at least three dedicated, full-time engineers located onsite.

The Manager - Offsite Safety Review and staff shall meet or exceed the qualifications described in Section 4.7 of ANS 3.1 - 1981 and shall be guided by the provisions for independent review described in Section 4.3 of ANSI N18.7 - 1976 (ANS 3.2).

The Offsite Safety Review staff shall generally possess experience and competence in the areas listed in Section 6.5.2.4.1. A system of qualified reviewers from other technical organizations shall be utilized to augment expertise in the disciplines of Section 6.5.2.4.1, where appropriate. Such qualified reviewers shall meet the same qualification requirements as the Offsite Safety Review staff, and shall not have been involved with performance of the original work.

The Onsite Safety Review Engineer and staff shall meet or exceed the qualifications described in Section 4.4 of ANS 3.1 - 1981.

CONSULTANTS

6.5.2.3 Consultants or other technical experts shall be utilized by NSR to the extent necessary as determined by the General Manager - Nuclear Safety Review.

6.5.2.4 OFFSITE SAFETY REVIEW (OSR)

FUNCTION

6.5.2.4.1 The OSR organization shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,

b. Nuclear engineering,

c. Chemistry and radiochemistry,

d. Metallurgy,

e. Instrumentation and control,

f. Radiological safety,

g. Mechanical engineering,

h. Electrical engineering

i. Quality assurance

j. Nondestructive testing

k. Emergency preparedness

REVIEW

6.5.2.4.2 The OSR shall review:

a. The safety evaluations for changes to procedures, equipment, or systems; and tests or experiments completed under the provision of 10 CFR 50.59 to verify that such actions did not constitute an unreviewed safety question;

b. Proposed changes to procedures, equipment, or systems and tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

c. Proposed changes to Technical Specifications or this Operating License;

d. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

e. Significant operating abnormalities or deviations from normal and expected performance of facility equipment that affect nuclear safety;

464

f. All REPORTABLE EVENTS.

g. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

h. Reports and meeting minutes of the SORC.

AUDITS

6.5.2.4.3 Audits of facility activities shall be performed under the cognizance of the OSR. These audits shall encompass:

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months;

b. The performance, training and qualifications of the entire facility staff at least once per 12 months;

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems, or method of operation that affect nuclear safety, at least once per 6 months;

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months;

e. The Facility Emergency Plan and implementing procedures at least once per 12 months;

f. The Facility Security Plan and implementing procedures at least once per 12 months;

g. Any other area of facility operation considered appropriate by the General Manager - Nuclear Safety Review or the Vice President - Nuclear;

h. The facility Fire Protection Program and the implementing procedures at least once per 24 months;

i. The fire protection and loss prevention program implementation at least once per 12 months utilizing either a qualified off-site licensee fire protection engineer(s) or an outside independent fire protection consultant. An outside independent fire protection consultant shall be utilized at least once per 36 months; and

j. The radiological environmental monitoring program and the results thereof at least once per 12 months.

k. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months;

465

l. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months; and,

m. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring at least once per 12 months.

The above audits shall be conducted by the Nuclear Quality Assurance Department or an independent consultant. Audit plans and final audit reports shall be reviewed by the OSR prior to issuance.

RECORDS

6.5.2.4.4 Records of OSR activities shall be maintained. Reports of reviews and audits shall be prepared and distributed as indicated below:

a. The results of reviews performed pursuant to Section 6.5.2.4.2 shall be reported to the Vice President - Nuclear at least monthly.

b. Audit reports prepared pursuant to Specification 6.5.2.4.3 shall be forwarded by the auditing organization to the Vice President - Nuclear and to the management positions responsible for the areas audited (1) within 30 days after completion of the audit for those audits conducted by the Nuclear Quality Assurance Department, and (2) within 60 days after completion of the audit for those audits conducted by an independent consultant.

6.5.2.5 ONSITE SAFETY REVIEW GROUP (SRG)

6.5.2.5.1 The SRG shall function to provide: the review of plant design and operating experience for potential opportunities to improve plant safety; evaluation of plant operations and maintenance activities; and advice to management on the overall quality and safety of plant operations.

The SRG shall make recommendations for revised procedures, equipment modifications, or other means of improving plant safety to appropriate station/corporate management.

RESPONSIBILITIES

6.5.2.5.2 The SRG shall be responsible for:

a. Review of selected plant operating characteristics, NRC issuances, industry advisories, and other appropriate sources of plant design and operating experience information which may indicate areas for improving plant safety.

b. Review of selected facility features, equipment, and systems.

c. Review of selected procedures and plant activities including maintenance, modification, operational problems, and operational analysis.

d. Surveillance of selected plant operations and maintenance activities to provide independent verification* that they are performed correctly and that human errors are reduced to as low as reasonably achievable.

AUTHORITY

6.5.2.6 NSR shall report to and advise the Vice President - Nuclear on those areas of responsibility specified in Sections 6.5.2.4 and 6.5.2.5.

6.5.3 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.3.1 All programs and procedures required by Technical Specification 6.8 and changes thereto, and any other proposed procedures or changes thereto, which affect plant nuclear safety as determined by the General Manager - Hope Creek Operations, other than editorial or typographical changes, shall be reviewed as follows:

PROCEDURE RELATED DOCUMENTS

6.5.3.2 Procedures, Programs and changes thereto shall be reviewed as follows:

a. Each newly created procedure, program or change thereto shall be independently reviewed by an individual knowledgeable in the area affected other than the individual who prepared the procedure, program or procedure change, but who may be from the same organization as the individual/group which prepared the procedure or procedure change. Procedures other than Station Administrative procedures will be approved by the appropriate station Department Manager or by the Assistant General Manager - Hope Creek Operations. Each station Department Manager shall be responsible for a pre-designated class of procedures. The General Manager - Hope Creek Operations shall approve Station Administrative Procedures, Security Plan implementing procedures and Emergency Plan implementing procedures.

b. On-the-spot changes to procedures which clearly do not change the intent of the approved procedures shall be approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License. Revisions to procedures which may involve a change in intent of the approved procedures, shall be reviewed in accordance with Section 6.5.3.2.a above.

c. Individuals responsible for reviews performed in accordance with item 6.5.3.2.a above shall be approved by the SORC Chairman and designated as a Station Qualified Reviewer. A system of Station Qualified Reviewers shall be maintained by the SORC Chairman. Each review shall include a written determination of whether or not additional cross-disciplinary review is necessary. If deemed necessary, such review shall be performed by the appropriate designated review personnel. The Station Qualified Reviewers shall meet or exceed the qualifications described in Section 4.4 of ANS 3.1, 1981.

---

*Not responsible for sign-off function.

d. If the Department Manager determines that the documents involved contain significant safety issues, the documents shall be forwarded for SORC review and also to NSR for an independent review to determine whether or not an unreviewed safety question is involved. Pursuant to 10 CFR 50.59, NRC approval of items involving unreviewed safety questions or requiring Technical Specification changes shall be obtained prior to implementation.

NON-PROCEDURE RELATED DOCUMENTS

6.5.3.3 Tests or experiments, and changes to equipment or systems shall be forwarded for SORC review and also to NSR for an independent review to determine whether or not an unreviewed safety question is involved. The results of NSR reviews will be provided to SORC. Recommendations for approval are made by SORC to the General Manager - Hope Creek Operations. Pursuant to 10 CFR 50.59, NRC approval of items involving unreviewed safety questions or requiring Technical Specification changes shall be obtained prior to implementation.

RECORDS

6.5.3.4 Written records of reviews performed in accordance with item 6.5.3.2a above, including recommendations for approval or disapproval, shall be maintained. Copies shall be provided to the General Manager - Hope Creek Operations, SORC, NSR, and/or NRC as necessary when their reviews are required.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified pursuant to the requirements of Section 50.72 to 10 CFR Part 50 and a report submittal pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the SORC, and the results of this review shall be submitted to the NSR and the Vice President - Nuclear.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President - Nuclear and the General Manager - NSR shall be notified within 24 hours.

468

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon unit components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the General Manager - Nuclear Safety Review and the Vice President - Nuclear within 30 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978.

b. The applicable procedures required to implement the requirements of NUREG-0737 and supplements thereto.

c. Refueling operations.

d. Surveillance and test activities of safety-related equipment.

e. Security Plan implementation.

f. Emergency Plan implementation.

g. Fire Protection Program implementation.

h. PROCESS CONTROL PROGRAM implementation.

i. OFFSITE DOSE CALCULATION MANUAL implementation.

j. Quality Assurance Program for effluent and environment monitoring.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed and approved in accordance with specification 6.5.1.6 or 6.5.3, as appropriate, prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 On-the-Spot changes to procedures of Specification 6.8.1 may be made provided:

a. The intent of the original procedure is not altered;

b. The change is approved by two members of the unit management staff, at least one of whom holds a Senior Reactor Operator license on the unit affected; and

AVAILABLE

PROCEDURES AND PROGRAMS (Continued)

c. The change is documented and receives the same level of review and approval as the original procedure under Specification 6.5.3.2a within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the HPCI, CS, RHR, RCIC, Containment Hydrogen Recombiner, H_2/O_2 analyzer, Post-Accident Sampling, Control Rod Drive Hydraulic (Scram Discharge portion) systems. The program shall include the following:

1. Preventive maintenance and periodic visual inspection requirements, and
2. A service pressure leak test for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,
2. Procedures for monitoring, and
3. Provisions for maintenance of sampling and analysis equipment.

c. Post-accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1. Training of personnel,
2. Procedures for sampling and analysis, and
3. Provisions for maintenance of sampling and analysis equipment.

470

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

6.9.1.2 The startup report shall address each of the tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the startup report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS[*]

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions[**] (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling). The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole-body dose received from external sources should be assigned to specific major work functions; and

---

[*] A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

[**] This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

b. Documentation of all challenges to main steamline safety/relief valves.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.6 Routine radiological environmental operating reporting covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The annual radiological environmental operating reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

The annual radiological environmental operating reports shall include summarized and tabulated results in the format of Regulatory Guide 4.8, December 1975 of all radiological environmental samples taken during the report period. Deviations from the sampling program identified in Technical Specification 3.12.1 shall be reported. In the event that some results are not available for inclusion with the reports, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; a map of all sampling locations keyed to a table giving distances and directions from one reactor; and the results of licensee participation in the Interlaboratory Comparison Program, as required by Specification 3.12.3.

The report shall also include the results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.5. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (Continued)

after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.1.7 Routine radioactive release reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The radioactive effluent release reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The radioactive effluent release report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing of wind speed, wind direction, and atmospheric stability, and precipitation (if measured) on magnetic tape, or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1.1-1) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The historical annual average meteorology or the meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM). The Semiannual Radioactive Effluent Release Report shall identify those radiological environmental sample parameters and locations where it is not possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In addition, the cause of the unavailability of samples for the pathway and the new location(s) for obtaining replacement samples should be identified. The report should also include a revised figure(s) and table(s) for the ODCM reflecting the new location(s).

413

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The radioactive effluent release report to be submitted within 60 days after January 1, of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous 12 consecutive months to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1.

The radioactive effluents release shall include the following information for each class of solid waste (as defined by 10 CFR 61) shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclide (specify whether determined by measurement or estimate),

d. Type of waste (e.g., spent resin, compact dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent (e.g., cement, urea formaldehyde).

The radioactive effluent release reports shall include unplanned releases from the site to the UNRESTRICTED AREA of radioactive materials in gaseous and liquid effluents on a quarterly basis.

The radioactive effluent release reports shall include any changes to the PROCESS CONTROL PROGRAM (PCP), OFFSITE DOSE CALCULATION MANUAL (ODCM) or radioactive waste systems made during the reporting period.

MONTHLY OPERATING REPORTS

6.9.1.8 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office no later than the 15th of each month following the calendar month covered by the report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

6.9.3 Violations of the requirements of the fire protection program described in the Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be reported to the Regional Administrator of the Regional Office of NRC via the Licensee Event Report System within 30 days.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

SPECIAL REPORTS

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.3 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 5.7.1-1.

RECORD RETENTION (Continued)

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of inservice inspections performed pursuant to these Technical Specifications.

i. Records of quality assurance activities required by the Quality Assurance Program.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the SORC and the NSR activities.

l. Records of the snubber service life monitoring pursuant to Technical Specification 4.7.5.

m. Records of analyses required by the radiological environmental monitoring program which would permit evaluation of the accuracy of the analyses at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr* shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)**. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

---

*Measurement made at 18 inches from source of radioactivity.

**Radiation protection personnel or personnel escorted by radiation protection personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

476

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.

c. A radiation protection qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Radiation Protection Supervisor in the RWP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose greater than 1000 mrem* shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Senior Nuclear Shift Supervisor on duty and/or the radiation protection supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose in excess of 1000 mrem* that are located within large areas, such as the containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted, and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, continuous surveillance direct or remote (such as use of closed circuit TV cameras), may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

---

*Measurement made at 18 inches from source of radioactivity.

PROCESS CONTROL PROGRAM (PCP) (Continued)

a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

c. Documentation of the fact that the change has been reviewed and found acceptable by the SORC.

2. Shall become effective upon review and acceptance by the SORC.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determination; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the SORC.

2. Shall become effective upon review and acceptance by the SORC.

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS

6.15.1 Licensee initiated major changes to the radioactive waste system (liquid, gaseous and solid):

1. Shall be reported to the Commission in the FSAR for the period in which the evaluation was reviewed by SORC. The discussion of each changes shall contain:

478

MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS (Continued)

a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR Part 50.59;

b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

d. An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

e. An evaluation of the change, which shows the expected maximum exposures to individual in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;

f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

g. An estimate of the exposure to plant operating personnel as a result of the change; and

h. Documentation of the fact that the change was reviewed and found acceptable by the SORC.

2. Shall become effective upon review and acceptance by the SORC.

479

NRC FORM 335
(2-84)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

1. REPORT NUMBER *(Assigned by TIDC, add Vol. No., if any)*
NUREG-1202

TITLE AND SUBTITLE
Technical Specifications for Hope Creek Generating Station

3. LEAVE BLANK

4. DATE REPORT COMPLETED
MONTH July YEAR 1986

5. AUTHOR(S)

6. DATE REPORT ISSUED
MONTH July YEAR 1986

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Division of BWR Licensing
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Same as 7. above

11a. TYPE OF REPORT
Technical Report

b. PERIOD COVERED *(Inclusive dates)*

12. SUPPLEMENTARY NOTES
Pertains to Docket No. 50-354

13. ABSTRACT *(200 words or less)*

The Hope Creek Generating Station Technical Specifications were prepared by the U. S. Nuclear Regulatory Commission to set forth the limits, operating conditions, and other requirements applicable to a nuclear reactor facility as set forth in Section 50.36 of 10 CFR 50 for the protection of the health and safety of the public.

14. DOCUMENT ANALYSIS – a. KEYWORDS/DESCRIPTORS

b. IDENTIFIERS/OPEN ENDED TERMS

15. AVAILABILITY STATEMENT
Unlimited

16. SECURITY CLASSIFICATION
(This page)
Unclassified
(This report)
Unclassified

17. NUMBER OF PAGES

18. PRICE

480

BEST COPY AVAILABLE